Nuclear Power Plant Innovation for the 1990s:

A Preliminary Assessment

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1. EXECUTIVE SUMMARY

The purpose of this report is two-fold:

(1) To present a preliminary assessment of the possible role of nuclear power plant design innovations in enhancing the attractiveness of the nuclear energy option to U.S. electric utilities; and

(2) To suggest some promising avenues for further technological development, and to define a role for MIT in the context of these proposed efforts.

The main findings are as follows:

A. The Need for Nuclear Power and the Role of Nuclear Power Plant Design Innovations

(1) It is highly unlikely that any orders for new nuclear plants will be placed by U.S. utilities before the end of the decade. Several factors are responsible, including a general lack of need for additional power and the plethora of uncertainties surrounding the performance of the current generation of nuclear plants. Nevertheless, there is a broad range of plausible circumstances -- even including those in which electricity demand grows a good deal more slowly than the latest mid-range forecasts of the government and the electric power industry -- under which nuclear energy

could once again be called on to supply an important fraction of the nation's new and replacement electricity supplies by the turn of the century.

(2) It is possible to conceive of circumstances in which new nuclear plants would assume a dominant role; this could occur, for example, if concerns about the environmental impacts of coal combustion were to escalate much above the present level. It is also possible that a future nuclear contribution could be ruled out entirely, for example if electricity demand were to grow at a rate of 1% per year or less, or if one of the nation's 120 or so nuclear plants now in operation or under construction were to undergo a major accident.

(3) On balance, we believe that the likelihood of an important national role for new nuclear power plants towards the end of the 1990s and for some time thereafter is sufficiently high that the electric power industry, the nuclear supply industry, and the government should explore carefully the conditions that would foster a resumption of nuclear ordering beginning in the early 1990s, and that industry and government, both separately and in partnership, should take steps to ensure that these conditions will be met at the appropriate time.

(4) Several of the more important reasons why utilities are not presently ordering nuclear power plants are beyond the ability of the nuclear industry and the nuclear-related government agencies

directly to control. This is clearly true of the sharp drop in the rate of electricity demand growth, and of the present high cost of capital. Other problems are less obviously out of reach, but still revolve around issues -- political, constitutional, social -- which transcend the particular technological characteristics of nuclear power plants: for example, the impact of the separate and overlapping jurisdictions of the states and the federal government on the climate of economic and environmental regulation; the rights of individuals and interest groups to intervene in regulatory proceedings, and the effect on the predictability of the regulatory process, and the delay and uncertainty introduced by litigation and judicial review.

(5) On the other hand, some of the most important reasons for electric utilities not to build new nuclear plants are more directly susceptible to the corrective efforts of industry and government. Many of the worst problems to befall the nuclear industry have occurred as a result of its own management failures; the best-known examples include mistakes in the execution of designs, mismanagement of construction projects, and poor operating practices. Upgrading the quality of the management of nuclear plant technology is of vital importance to the future of the nuclear option.

(6) The importance of the nuclear industry's efforts to improve its management performance cannot be overstated, and such efforts

demand the highest priority. Nevertheless, doing better with the technology that is already available is not the only possible goal in this regard. Management effectiveness is strongly coupled to the nature of the technology itself. A fundamental cause of problems in nuclear construction and operation, for example, is the extraordinary complexity of contemporary light water reactor plant designs. Potentially large opportunities exist for nuclear power plant design innovations which would reduce the demands placed on their builders and operators, or which would enhance the attractiveness of nuclear power to utilities in other ways. Relatively little attention is being paid to these opportunities at present.

(7) We do not speculate on whether the realization of such innovations is a necessary condition for the revitalization of the nuclear option. However, we believe that technological changes are in prospect which could substantially improve the prospects for a resumption of nuclear ordering in the 1990s, which are within reach within the time available, but which are unlikely to be realized given the current direction of industry and government policies.

(8) The traditional view of nuclear power plant innovation, formulated during a period of rapid expansion of the nuclear industry, has been that competition among alternative reactor

types in the long run will be decided primarily by the criterion of uranium utilization efficiency, and that the key economic variable in this competition is the price of natural uranium. The primary objective of innovation efforts, according to this view, is to ensure that nuclear plant systems with suitably conservative uranium consumption characteristics are available as required by depletion-driven uranium price increases. However, for the foreseeable future, the market position of nuclear plants in the U.S. relative to each other and to alternative sources of electric power will be primarily determined by other factors. Among the most important of these are the plant capital cost, the construction lead-time, the operating reliability, and the perceived and actual risks to public safety and to utility investors associated with the plants. The changed economic and political outlook for nuclear power in the United States demands a fundamental reexamination of the premises and goals of the nation's nuclear power plant innovation efforts.

B. The Options

Two technological paths are available for the pursuit of these revised goals: one is based on evolutionary improvements to existing light water reactor designs, and the other involves radically different design concepts. Because of the large amount of experience accumulated with light water reactors and the very large investment already made in this technology, the electric power industry will be naturally inclined towards the former route. These are not in themselves sufficient grounds to rule out the latter, however.

(1) The present generation of LWRs has been aptly described as the product of a 'band-aid' approach to design, especially with regard to safety, with safety-related systems added over time to meet safety requirements imposed after completion of the initial design of the plant. No plant today is the result of a systematic, optimized approach to the satisfaction of current safety requirements. Neither have the safety requirements themselves been rationalized to resolve inconsistencies and to establish a hierarchy among the sometimes conflicting goals that have evolved over time. The present hiatus in nuclear ordering provides an opportunity to undertake a major rationalization both of LWR plant safety criteria and of plant designs. A key goal of these efforts would be to achieve substantial design simplification. In our view, the current activities of the Nuclear Regulatory Commission and the nuclear reactor suppliers fall well short of what could be achieved in these areas.

(2) Of the other design concepts, a fully modularized system of small pebble-bed high temperature gas reactors appears to offer significant promise, when all relevant design goals are considered. In such plants, the economies of scale of large individual reactors are exchanged for the economies of serial production and shop fabrication of small standardized units.

Small, pebble-bed reactors exhibit a high level of inherent safety, simplicity of design, a potentially high capacity factor and a high thermal efficiency. Operation is relatively straightforward and quite readily suited to full automation, and dose levels to operating and maintenance personnel are low relative to other systems. Compared with large individual plants, modular arrays of small units would permit utilities to make smaller incremental investments in new generating capacity, and to match capacity expansions more closely to the availability of financing and to load growth. The combination of on-line refueling and moderately redundant modular arrays promises high overall plant availability, as well as a much lower probability of full plant forced outages than with large individual units. Although there are unresolved engineering and economic issues associated with this concept, resolution of these questions appears to be feasible within a five year time frame.

(3) The PIUS reactor system, a radically redesigned LWR concept, has the advantage of being able to draw substantially on available LWR technology, and appears to be significantly less likely to undergo severe core damage than LWRs of conventional design. However, whether it can meet utility requirements in other key areas, including reliability, maintainability, and adaptability to downsizing and accelerated construction, remains to be seen.

(4) The CANDU-type reactor is by far the most highly developed of the heavy water-moderated systems for central station power generation, and has compiled an impressive operating record in Canada. However, we doubt that the CANDU system, or any evolutionary extension of its essential features, offers the prospect of a large enough advance beyond the performance of current LWRs in ways important to the concerns of this study to warrant its introduction into the United States.

(5) Regarding the LMFBR, we believe that the basic design concept is potentially more responsive to expected conditions in the electric utility industry during the next few decades than present designs would indicate. However, compared with the other options, the available technology base and operating experience is narrower and less advanced, and could not support commercialization in the time frame of interest here.

C. The Innovation Process

(1) Present conditions in the nuclear power industry are not favorable to the active pursuit of technological innovation in the nuclear power plant field. For different reasons, the electric utility industry, the nuclear plant suppliers, and the government are all reluctant to undertake major new initiatives at this stage.

(2) The problems are deep-rooted, and without a basic change in assumptions and policies no amount of creativity in organizational restructuring will be sufficient to stimulate more vigorous efforts to develop innovative power plant system designs for the 1990s.

(3) The most important requirement is for the electric utility industry to recognize its own pre-eminent role in ensuring the timely availability of the technology which will enable it to sustain and improve the quality of its service. Without a clear financial commitment by the utilities, neither the traditional suppliers of nuclear power plant nor the federal government will be prepared to increase their own efforts in this area. Further, the the difficulties encountered in past nuclear power plant innovation efforts amply demonstrate the importance of a technically well-informed user and of good technical communication between the user and the supplier of the plants. Without the active technical participation of the utilities in the formulation of design specifications and, where applicable, in the construction and operation of prototypes, future nuclear power plant innovation efforts appear much less likely to succeed.

D. The Role of MIT

(1) Developing and implementing major nuclear power plant design changes would require a very large commitment of technical, financial and probably also political resources over an

extended period. No university research effort can make more than a minor contribution on this scale. But in the shorter run, however, the role of a small, independent research group such as exists at MIT may be more important. By being prepared to take the lead in addressing some of the more difficult, speculative, or sensitive design-related issues which government and industry organizations are unable or unwilling to act on at this stage, such a group could play a key role in catalyzing the much larger scale effort that would be required for full implementation. The rationalization of LWR plant safety requirements is one area where the MIT group could conceivably play such a role. Other areas include system simplification through probabilistic and other methods, and the application of advanced control technology.

(2) A new program of research at MIT is proposed in the area of advanced nuclear power plant design and supporting research and development for systems targeted for commercial introduction in the 1990s. The program will be centered on engineering development projects in two areas -- evolutionary improvements in LWR designs, and small, modular high temperature gas reactor systems; it will also include supporting engineering and social science research and policy studies. The program will be based in the Nuclear Engineering department, but will draw on the full range of relevant interests and experience at the Institute.

(3) LWR Innovation. It is proposed to form a research consortium in association with a group of leading utilities with the goal of developing a rationalized design specification for an advanced LWR plant tailored to the projected needs of the utilities in the 1990s. The duration of the project would be approximately 5 years. With four to eight utility participants, each contributing \$200,000/yr, the MIT research budget would amount to \$0.8-1.6 million/yr. Each utility would also be asked assign two technical staff members to the project.

(4) Small modular HTGR systems development. A new MIT initiative is proposed in the area of small HTGR systems. The main thrust of this project will be in the area of modular reactor systems studies. The ultimate target funding level for the project is \$0.5-1 million/yr. Utility sponsors will be sought from the membership of Gas Cooled Reactor Associates. The goal of the project is to determine whether small, modular HTGRs should be promoted as next generation power systems. The first phase of the project would be completed in 3-5 years.

2. INTRODUCTION

This report examines the role that nuclear power plant design innovations might play in making nuclear energy a more attractive generating option for U.S. electric utilities in the 1990s and beyond. It addresses both the commercial potential of such innovations and the feasibility of bringing them to market.

Even under normal circumstances -- given the relative youth of the civil nuclear industry -- the rapid accumulation of nuclear plant construction and operating experience over the last few years would provide a rich source of ideas for future design refinements and modifications.* As it is, several recent developments, including some which threaten the survival of the industry, have focused new attention on the possibilities of technological innovation, while at the same time creating an environment in which the continuation of the innovation process itself is in jeopardy. These developments include the following:

- American electric utilities stopped ordering nuclear power plants in the mid-1970s, and a resumption of ordering is not presently in prospect. Many factors have contributed to this situation, some of them closely related to the technical characteristics of the plants themselves. If further ordering is ever to take place, utility managements must be convinced that

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* It is noteworthy that as much LWR operating experience (measured in reactor-years of operation) has been gained since 1979 as in the preceding twenty years. For construction experience the picture is similar.

the nuclear option is economic, safe, and politically acceptable, and hence competitive with alternative sources of electric power. This is largely not the case today. Attempts to restore the competitiveness of the nuclear option will involve an array of institutional measures aimed at rationalizing plant construction and operating practices and regulatory procedures. It is also appropriate to ask to what extent changes in the design of these plants can affect utility attitudes, and how such changes might be implemented.

- The present hiatus in nuclear ordering in principle increases the number of design options which could potentially be made available commercially in time for the next round of orders, should there be one. It is now generally expected that no new orders will be placed by U.S. utilities before the end of the decade, at the earliest. Though this outlook poses difficult problems of adjustment for the nuclear supply industry, and seriously complicates the innovation process, it also opens an opportunity for the exploration of a broader array of options than might have been considered for the next generation of designs in a growing market.

- The sharp decline in the expected growth rate of nuclear power plant capacity coupled with the more promising outlook for the availability of reasonably-priced uranium resources means that uranium conservation need no longer be the overriding objective of reactor designers during the next few decades. The fuel supply outlook is further enhanced by the shift towards higher burnup fuel and the favorable prospects for advanced enrichment technologies. Together, these developments open a commercial 'window of opportunity' for design innovations which may not offer any major improvements in the efficiency of uranium use over light water reactors of conventional design, but whose advantages lie in other, increasingly important directions. In previous years, the case for such innovations -- especially those demanding large initial investments -- would have been undermined by expectations of early obsolescence in the face of rapidly rising uranium prices and the prompt commercial introduction of the fast breeder reactor.

- The changing structure of the world nuclear power plant industry raises new questions regarding U.S. innovation strategy, as the number of nations at the forefront of reactor technology grows. In the light water reactor area, several European nations and more recently Japan have joined American firms at the technological frontier; West Germany is establishing a strong position in advanced high temperature gas reactor technology; and the American liquid metal fast breeder reactor development program, though still the world's largest, may no longer be the most advanced, at least in terms of experience with large-scale operating systems. Moreover, the structure of the nuclear power plant industry is increasingly becoming integrated across national boundaries. In this new environment, the importation of reactor technology into the U.S. is for the first time becoming a plausible option. The question arises as to whether and in what circumstances it would be an attractive one. Other questions concern the export competitiveness of the American nuclear industry. If, as seems inevitable, the U.S. must forego its traditional dominance across the full spectrum of civil nuclear technologies, in which specific segments of the industry should it seek to retain technological and commercial leadership? Also, under what circumstances would an innovation strategy which sought to 'leapfrog' the current efforts of its international competitors be appropriate?

Unfortunately, the U.S. nuclear power plant industry at present appears not to have sufficient vitality to respond effectively to these challenges. The interest of the electric utilities in innovations is largely confined to developments which could be applied at small cost to plants already in operation or under construction. With a substantial surplus of generating capacity at present, faltering demand growth, and a record of vastly underestimating the difficulty of introducing the current generation of nuclear power plants, the utilities see few incentives to promote the development of another generation of systems. Moreover, to the extent that such developments would reflect negatively on public or regulatory perceptions of their existing nuclear investments, the utilities will actually have a disincentive to undertake such an exploration.

In turn, the reactor vendors, whose previous nuclear plant design and manufacturing activities have generally been unprofitable, whose investments in the current product line are yet to be fully amortized, and whose market projections beyond the next several years are highly uncertain, are not contemplating the allocation of a large amount of risk capital to the development of advanced reactor designs at present. The major design activity in which the two leading U.S. light-water reactor vendors, Westinghouse and General Electric, are currently engaged is being conducted in cooperation with their nuclear reactor manufacturing licensees in Japan, with the Japanese government, utilities and manufacturers providing most of the funds.* The goal of these two projects is to develop advanced light water reactor designs for commercial introduction into the Japanese electric utility market by the mid-1980s. (The first of these plants would then enter service in the early 1990s.) At present, the two American vendors are making little if any effort to explore the possibilities of introducing these advanced systems into the U.S. market. None of the U.S. vendors is engaged in any other advanced design activity of comparable magnitude.

Finally, the government, which for many years has concentrated its nuclear power plant innovation efforts almost exclusively on the development of the liquid metal cooled fast breeder reactor, now confronts an economic environment in which commercial breeder deployment is a much more distant prospect, but so far has been unable to reach a consensus on an innovation strategy appropriate for these new conditions.

This report has two purposes:

(1) To present a preliminary assessment of the role of nuclear power plant design innovations in enhancing the attractiveness of the nuclear energy option to U.S. electric utilities; and

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* The so-called ABWR (advanced boiling water reactor) project involves General Electric, Toshiba, Hitachi, and a group of BWR-owning Japanese electric utilities led by Tokyo Electric Power Company. The participants in the APWR (advanced pressurized water reactor) project are Westinghouse, Mitsubishi Heavy Industries, and a PWR utility group led by Kansai Electric Power. The combined cost of these two projects is reportedly on the order of $200-250 million (Nucleonics Week, 3 March 1982, p. 4; Oriental Economist, January 1982, p. 12).

(2) To suggest some promising avenues for further development, and to define a role for M.I.T. in the context of these proposed efforts.

We approach these issues with the perspective of the electric utilities uppermost in mind. Unless the utilities -- or a subset of them -- are persuaded of the possible need for advanced nuclear power plant systems and, equally important, actively participate in defining the requirements to be met by those systems, it is highly unlikely that any effort to implement significant design innovations will succeed. From the utilities' perspective, the central issue is whether an advanced nuclear power plant system could more closely approach the (often conflicting) demands of ratepayers, stockholders, regulators, and other interested constituencies than could the most attractive of the available electricity supply alternatives.

Other perspectives are also important, however. For current or prospective private sector suppliers of advanced power plant technology the question is whether the potential demand for such a product, assuming it can be engineered and manufactured at a profit, would be large enough to justify the costs required to develop it. And for the government, the question is whether there are advanced nuclear technologies which are capable of supplying electric power more cheaply and/or with less environmental insult than the alternatives (including conventional nuclear systems) but which are unlikely to materialize if reliance is placed exclusively on existing market mechanisms to make them available.

It is important to emphasize what is not addressed in this report. First, we do not analyse in detail the range of possible design modifications to existing plants or to those now under construction. We recognize that the opportunities here are considerable, that the benefits are more immediate, and that such improvements could go some way towards restoring the competitiveness of the nuclear option. (We note, for example, that increasing the capacity factor of the operating nuclear plants from the current average of about 60% to the original design objective of 80% would not only reduce the cost of nuclear electricity by a substantial margin but would also effectively add 15,000 MWe of capacity to the national electricity supply system.) However, the economic incentives to pursue such improvements are reasonably clear, and much work is already underway in this area. In contrast, for the reasons already mentioned the possible outlines of the next generation of nuclear plant designs are receiving relatively little attention, yet the longer-term benefits are potentially large.

Similarly, it should be noted that while we will not primarily be concerned with technological innovations in the remainder of the nuclear fuel cycle, there are, nevertheless, important opportunities for advance here, and in at least one case -- the management and disposal of high-level nuclear wastes -- further progress is almost certainly a prerequisite for future nuclear power plant ordering.

Finally, our focus is almost exclusively on the domestic electric utility industry: the international prospects for advanced nuclear power plant technologies will not be addressed explicitly. We recognize that the expectations of government and industry regarding export prospects may critically influence the scale of any private or public commitment to large-scale innovation in the U.S. However, in our view there is little chance that American industry could successfully market nuclear power plant innovations overseas without first having proven them commercially at home. More generally, unless the domestic nuclear energy option is revitalized, the international competitiveness of the U.S. nuclear industry will almost certainly continue to decline.* Consequently, we believe that it is appropriate to concentrate on the opportunities presented by the U.S. electric utility market, at least as a first step.

Choosing among alternatives: some initial considerations

Even in theory, there can be no single universally 'optimal' nuclear power plant design concept, given the broad range of conditions in which plants are built and operated and the uncertain evolution of these conditions over time. Ideally, sufficient technological diversity would be sustained to enable a close-to-optimal system to be available for each set of circumstances that is likely to arise. In practice, the virtues of diversity must be weighed against the benefits of standardization and the high costs of

* The present collaborations of Westinghouse and General Electric with their Japanese partners are not inconsistent with this view. The American firms are playing an important role as system designers inthese projects. But the Japanese industry will supply virtually all of the hardware for the commercial plants (as indeed is the case for the current generation of plants under construction in Japan), and in the longer run seems likely to take the lead in any efforts to export the advanced systems to third countries.

pursuing many alternatives in parallel. Obviously, both the choice of design objectives and the selection of the technological means for pursuing them are strongly conditioned by the existing base of technology and experience. To this extent, future developments are determined by past decisions. Nevertheless, there is still some flexibility in the choice of direction and pace of development. In these decisions, the key considerations are the suitability (and hence marketability) of the proposed innovations given the range of likely commercial environments, and the feasibility of making them commercially available in the first place.

Perhaps the most fundamental issue that is addressed in this report -- aside from the question of whether the nuclear option is worth preserving at all -- is whether a technological strategy of evolutionary improvements to existing light water reactor designs is to be preferred to one based on radically different design concepts. Strong opinions are held on both sides of this issue. Proponents of an evolutionary improvement strategy believe that the option of a major shift away from conventional light water reactor technology is unrealistic and illusory. They argue that it is wiser to draw on the great store of LWR experience in order to move incrementally towards an improved system than to forego this experience in favor of an unproven concept. Though the latter may offer advantages on paper, in practice it would almost certainly have to overcome a long stream of unexpected problems similar to that which has plagued LWR technology, and from which the latter is only now emerging. Moreover, they point out the great practical difficulty of changing the technological course of a very large, inertia-ridden industry which for over two decades has been so strongly oriented towards LWR systems.

Advocates of the alternative strategy suggest that an evolutionary approach towards LWR improvements by its very nature may be insufficient to address the fundamental problems that have arisen with the nuclear option. They argue that a more radical technological shift would not entail a completely new start, with all of the inevitable problems of technological learning lying in the future. They point out that a good deal of the existing LWR technology base is likely to be transferable, irrespective of the direction of the shift, and that at least in the case of heavy water moderated or graphite moderated, gas-cooled systems a large amount of directly relevant experience already exists both in the U.S. and overseas. It is also suggested that the present reluctance of the industry to contemplate technological change may be significantly weaker a decade or so from now. (In this view, the longer the duration of the present moratorium on nuclear ordering the easier it is to envisage a major departure from conventional light water reactor technology.) Some of those urging a more radical technological departure take a stronger view, holding that disaffection towards conventional LWR technology in the electric utility industry and among the general public may be so strong that only a major technological change could prompt a return to the nuclear option.

In an ideal economic world, the marketplace would be the ultimate arbiter of these opposing points of view. In practice, however, the high entry barriers and large risks and costs associated with major innovations in the electric power plant industry mean that product competition is highly imperfect, and choices made far in advance of commercialization will limit the number and nature of the options made available to the utility customer, and hence profoundly and irreversibly influence the direction of technological change. To this extent, predictive analysis must replace marketplace interactions as a means of choosing an appropriate course of action. This report is an attempt to move beyond the largely subjective arguments presented so far towards the more detailed, objective appraisals of the individual options which such an analysis requires.

3. THE FUTURE MARKET FOR NUCLEAR POWER PLANTS

We next consider the potential demand by U.S utilities for new nuclear power plants through the year 2010. The size of this market will provide an indication of whether the costs of developing significant changes in nuclear power plant designs would be recoverable. The choice of 2010 as an endpoint is somewhat arbitrary, but reflects a balance between the intrinsic need for a long term assessment and the practical need to avoid being overwhelmed by the uncertainties involved. The primary task of this section is to estimate the probable requirement for new baseload electricity supplies; if nuclear power is to regain competitiveness it will again primarily be as a baseload supplier, because of the high capital intensity and low fuel costs of nuclear plants relative to most conventional alternatives. At least until the end of the century, and possibly for a good deal longer, the major competition for new baseload applications will come from coal-fired plants, although in certain localities other sources (natural gas, geothermal, hydro) may play an important role.

3.1 Electricity demand projections

The size of the market for baseload power plants will be primarily determined by future trends in the demand for electricity. No attempt was made to develop independent electricity demand projections for this project. Instead, we reviewed a number of recent studies to establish a representative range of anticipated growth rates.

The key influences on electricity demand are the overall size of the economy; the aggregate price of energy; the relative prices of electricity and other forms of energy; and energy price elasticities in different sectors of the economy. These last are in turn strongly influenced by the prevailing cost of capital. In the longer run, structural changes in the composition of the gross national product induced by foreign economic competition, technological innovation, and changing social preferences may have a major impact. Large uncertainties remain regarding the contributions of each of these factors.

Several recent demand growth projections for the period 1981-2000 are summarized in Table 3.1. During this period, economic growth is generally the most important of the above-mentioned variables affecting electricity demand, and the GNP growth rates assumed for each of the projections are also shown in the Table. Over the longer term, the 'structural' factors acquire increased importance.

All the demand growth rate projections fall well below the average of 7% per year experienced in the decades prior to 1973, but the range is nevertheless large, from almost zero up to about 4% per year. The base-case projections by the government and the utility industry fall within a narrower range, from 2.5 to 3.2%/yr. A recent analysis of several low-demand projections suggests that these differ from the mainstream of government and industry forecasts primarily as a result of different assumptions about electricity prices and consumer price responses. Specifically, the low-demand studies assume electricity prices higher in relation to other

Table 3.1

Selected Recent U.S. Electricity Demand Growth Projections

	Projection Period	Electricity Demand Growth Rate (% year)	Assumed GNP Growth Rate (% year)
DOE/Electricity Policy Project (6/83)	1981-2000	2.5(1.1-3.8)+	2.6(2.3-3.0)
DOE/Energy Information Administration (2/81)	1981-2000	2.6(2.2-3.0)	2.5
EPRI/Starr and Searl (4/82)	1981-2000	3.2(2.6-3.7)	2.5(2.0-3.0)
Electrical World (9/82)	1981-2000	2.6	
CONAES/Demand and Conservation Panel (1979)*	1975-2010		
Scenario A		0.6(0.2-2.5)	2.0
Scenario B		1.3(0.7-3.0)	2.0
Scenario C		2.7(2.2-4.0)	2.0
Congressional Research Service (8/82)	1981-1995	2.9(1.8-3.9)	3.0(2.0-4.0)
Solar Energy Research Institute (1981)	1978-2000	0.2(-1.4-0.4)	2.6

+ Range indicates upper and lower bounds of projections

* CONAES Scenarios:

A: Aggressive conservation policy; four-fold increase in average delivered energy price over 1975 price
B: Moderate conservation policy; two-fold increase in average delivered energy price over 1975 price
C: Unchanged conservation policy; energy prices unchanged.

(continued)

Sources for Table 3.1

DOE/Electricity Policy Project:	Report of the Electricity Policy Project, The Future of Electric Power in America: Economic Supply for Economic Growth, Office of Policy, Planning and Analysis, U.S. Department of Energy, June 1983.
DOE/Energy Information Administration:	Annual Report to Congress, 17, Vol. 3, DOE/EIA-0173(81)/3 (Washington, D.C. 1981).
EPRI/Starr and Searle:	C. Starr and M.F. Searl, "U.S. Generating Capacity Requirements for Economic Growth, 1990-2000," Public Utilities Fortnightly, April 29, 1982, 17-24.
Electrical World:	33rd Annual Electrical Industry Forecast, Electrical World, September 1982, 79.
CONAES:	National Research Council, Alternative Energy Demand Futures to 2010, Committee on Nuclear and Alternative Energy Systems, Demand and Conservation Panel (Washington, D.C.: National Academy of Sciences, 1979).
Congressional Research Service:	A. Kaufman and K.K. Nelson, "Do We Really Need All Those Electric Plants?", Report No. 81-1475, U.S. Library of Congress, Congressional Research Service, August 1982.
Solar Energy Research Institute:	The SERI Solar/Conservation Study, A New Prosperity: Building a Sustainable Energy Future, Brick House Publishing, Andover, MA, 1981.

energy prices than do other studies, as well as a lower propensity on the part of consumers to substitute electricity for other energy forms.* The GNP growth and aggregate energy price behavior assumed in these studies were not radically different, however.

3.2 Other relevant factors affecting the need for baseload plants

The size of the baseload plant market will depend on several factors in addition to the overall rate of growth of electricity demand:

(1) Because of daily and seasonal demand fluctuations, only part of the new demand can be met economically by conventional baseload power plants. At present, 60-70% of the nation's electricity is generated by units on baseload duty.** Changes in the shape of the load duration curve are probable during the time frame of this study, but not even the direction is clear. Time-of-day pricing will tend to flatten the curve, with a consequent increase in the share of total generation captured by baseload plants. Improved grid interconnections between different regions will have a similar effect. On the other hand, increases in intermittent generation (wind, solar, etc.) will reduce the baseload share.

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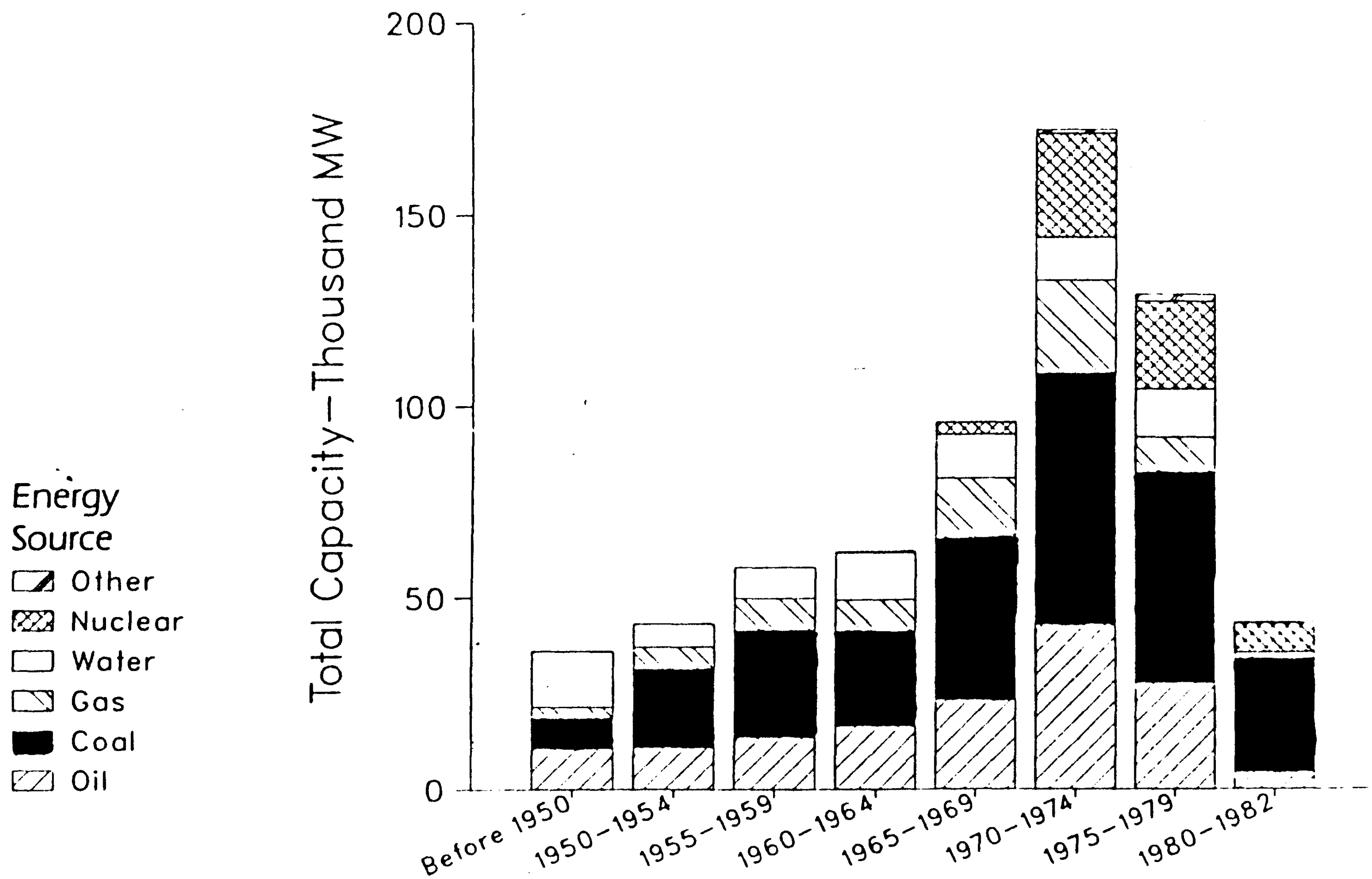
* Report of the Electricity Policy Project, The Future of Electric Power in America: Economic Supply for Economic Growth, Office of Policy, Planning and Analysis, U.S. Department of Energy, June 1983.

**Arthur Barstow, Northeast Utilities/New England Power Pool, private communication, June 26, 1983. Part of the uncertainty arises from the absence of a sharp distinction between baseload and intermediate load fossil plants.

(2) An additional requirement for new baseload plants will be created by the retirement of some older units and a decline in the productivity of others as they age. As Figure 3.1 shows, over 100,000 MWe of existing fossil steam plant, much of it baseload capacity, will be over 40 years old if still in service in the year 2000. In the present financial climate, many utilities are deferring plant retirements well beyond original design lifetimes to avoid major new capital investments. However, the incentives for early retirement of fossil baseload plants in some regions of the country are strong, and may be increasing. The cost of retrofitting some of the older coal plants to meet the more stringent emissions standards that are likely to go into effect shortly may be prohibitive. Also, substantial quantities of oil and gas are still being burned in baseload units. In 1982, oil and gas together accounted for about 21% of the nation's electricity supply; oil use by utilities averaged 0.85 million bbl/d (down from 1.5 million bbl/d in the mid-1970s), while gas use averaged 1.6 million bbl/d oil equivalent. Much of this was burned in baseload plants.* Although the economic incentives to displace

* The exact fraction is not easily determined. But both oil and gas use are concentrated in a few states, where they account for a large proportion of the electricity generated. In these states, the fraction used for baseload generation must therefore be high, and consequently so must the national fraction. (For a fuller discussion of this point, see: D. Bodansky, "A Strategy for Saving Oil," Journal of Contemporary Studies, Spring 1983, 84-85.) If the national fraction were 70%, say, the amount of baseload oil and gas capacity in service in 1982 would be approximately 88 GWe (assuming a 65% plant capacity factor).

Existing Capacity by Age and Energy Source



Source: Generating Units Reference File

Inventory of Power Plants in the United States
1981 Annual
Energy Information Administration

Fig. 3.1

baseload oil-fired capacity with new coal and nuclear plants have declined with the recent softening of world oil prices and the rapid increases in coal and nuclear capital costs, this will be at least partly offset by the increasing incentives for gas displacement as gas deregulation continues to take effect during the 1980s.

(3) An increase in the amount of industrially cogenerated electricity would reduce the need for new utility baseload capacity. At present, cogeneration provides only about 8% of industry's electricity supplies. Despite some very optimistic recent estimates of cogeneration potential, it is more likely that the actual future contribution will be quite modest. A careful recent analysis by Joskow at M.I.T. suggests that annual increments in cogeneration capacity for the next decade or so will probably be less than 1 GWe/yr, even if all economical opportunities are pursued -- a small fraction of total utility requirements for new capacity.*

(4) The need for new baseload capacity will also be affected by the introduction of new solar electric technologies (solar thermal

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* Paul Joskow, Industrial Cogeneration and Electricity Production in the United States, Studies in Energy and the Economy Discussion Paper No. 8, MIT-EL 81-061/WP, M.I.T., September 1981. A much more optimistic assessment of the cogeneration potential is given in the SERI Solar/Conservation Study, Building a Sustainable Energy Future (Brick House, Andover, MA, 1981). Here it is reported that exploitation of the cogeneration potential in six basic industries would displace almost 100 GWe of utility baseload capacity (p. 179).

electric, solar photovoltaic) and by the increased deployment of geothermal, biomass, hydroelectric, and wind capacity. Major new contributions from the latter sources are not expected, though in certain localities the incremental supply may be important. Regarding the new solar electric technologies, large uncertainties surround both the ultimate potential and the rate of market penetration. Because of the high costs of storage, these technologies will first become competitive on the grid in peak shaving and fuel saving applications. For small-scale, dispersed systems, such as photovoltaics, market penetration could occur quite rapidly once economic feasibility has been demonstrated, because of the short installation time relative to large central station plants. In the early stages, this would not affect utility baseload capacity requirements significantly. However, the generation of relatively large amounts of power from dispersed, intermittent solar electric systems could substantially alter the time patterns of electricity demand to be met by the electric utilities, and consequently affect their need for conventional baseload capacity.*

* At present, the cost of photovoltaic cell modules is still a factor of 15 higher than would be competitive for peaking or fuel saving applications. However, technological advances are in prospect which could make photovoltaics for these applications competitive at the margin by about the turn of the century. The development of economic baseload systems is a more distant prospect, as this would require further major reductions in module costs, along with major improvements in storage technology.

(5) The U.S. electricity grid is not a perfectly interconnected system. This reduces the overall efficiency of capacity utilization, thus increasing the need for new baseload plants.

3.3 The need for new baseload plants, 1990-2010

To investigate the combined effect of most of these factors on the size of the market for new baseload power plants, a very simple model of the U.S. electricity supply system was constructed. Three electricity demand growth scenarios were analysed:

Assumed growth rate (%/yr)

	1983-1990	1991-2000	2001-2010
HIGH	3	3.5	3
MID	2	2.5	2
LOW	1	1.5	1

Several other assumptions were made regarding the rate of capacity loss due to retirements and plant productivity decline; the fraction of new and replacement electricity demand that will be met by new 'conventional' central station baseload capacity (i.e. coal or nuclear) versus intermediate, peaking and dispersed baseload capacity; coal and nuclear plant capacity factors, and so on. A fuller description of these assumptions is given in the Appendix to this section.

Figure 3.2 shows the projected need for new baseload capacity for the three electricity demand scenarios. For each demand scenario, the projection more probably underestimates the actual need than exaggerates it. Two almost certainly conservative assumptions made in the analysis deserve special emphasis: first, oil and gas baseload generation is assumed to remain at its present level until 2010; and second, all of the coal units which the utilities are currently committed to build and all nuclear plants under construction except those that have been deferred indefinitely are assumed to be completed. Regarding the latter, it is probable that several more nuclear plants will be cancelled during the next two or three years, especially if electricity demand growth remains sluggish. Several coal plants may also be at risk.

According to Fig. 3.2, for the 'HIGH' demand growth scenario (3.3%/yr between now and 2000; 3%/yr thereafter) the first as yet unordered central station baseload plant will be required to enter service by 1993. Thereafter, new capacity will have to be added at a rate of approximately 17 GWe/yr. For the 'MID' case (2.3%/yr growth until 2000; 2%/yr afterwards), the target date will be deferred to around 1996, and subsequent capacity additions will be required at a rate of about 10 GWe/yr. And for the 'LOW' scenario (1.3%/yr until 2000; 1%/yr thereafter), the first new baseload plant will be required in about 2002, with subsequent additions at a rate of about 5 GWe/yr.*

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* These results have not been systematically compared with other recent projections. However, it is important to note that the criterion of need in this case is the adequacy of <u>economic baseload</u> capacity, while many other studies focus on the need for an adequate capacity reserve margin (typically taken as 20% above peak load) to ensure overall system reliability. Satisfaction of one criterion obviously need not imply satisfaction of the other.

Since even coal plants take on the order of a decade to complete, the likelihood of a shortfall in economic baseload capacity in the early 1990s seems high if electricity demand grows faster than 3% per year. In this case, the difference could be made up -- at higher cost -- by running intermediate and peaking units at a higher capacity factor or by deferring the retirement of old coal baseload plants. (As noted previously, however, the latter might well be precluded by the introduction of more stringent emission controls.) Some additional respite would be gained if the average capacity factor of existing coal and nuclear baseload plants could be increased above the 60% level assumed in Fig. 3.2. Figure 3.5 (the 'Maximum Delay' schedule) shows that by increasing the average retirement age of coal plants from 40 to 50 years and by increasing the capacity factor of all of the current generation of coal and nuclear plants to 65%, the need for new baseload capacity is delayed by 3 years in the 'HIGH' case and 4 years in the 'MID' case. The subsequent rate of capacity additions is essentially unchanged.

It is evident from Fig. 3.2 that even if electricity demand grows at less than 2% per year, new baseload plant completions will be required at a rate of several GWe per year starting in the late 1990s.

For each of the baseload capacity projections in Fig. 3.2, Figs. 3.3 and 3.4 show the potential market for new nuclear plants, assuming nuclear shares of the total baseload market of 50% and 25% respectively. The actual market share gained by nuclear will obviously depend on its competitiveness with coal in various parts of the country.

Fig. 3.2
NEED FOR NEW BASELOAD PLANTS: REFERENCE SCHEDULE

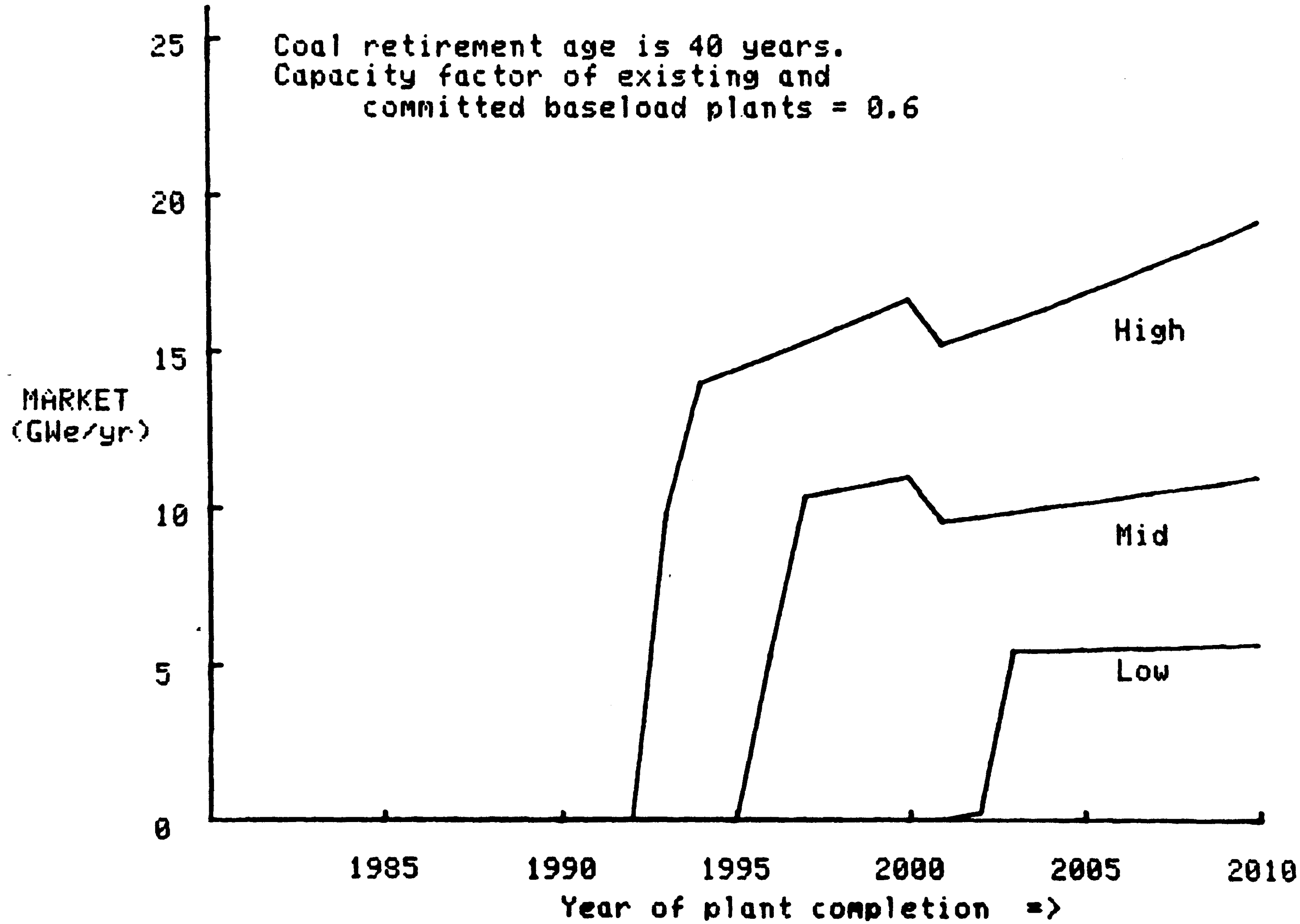


Fig. 3.3
NEW NUCLEAR PLANT MARKET: REFERENCE CASE
Nuclear share of baseload market is 0.50

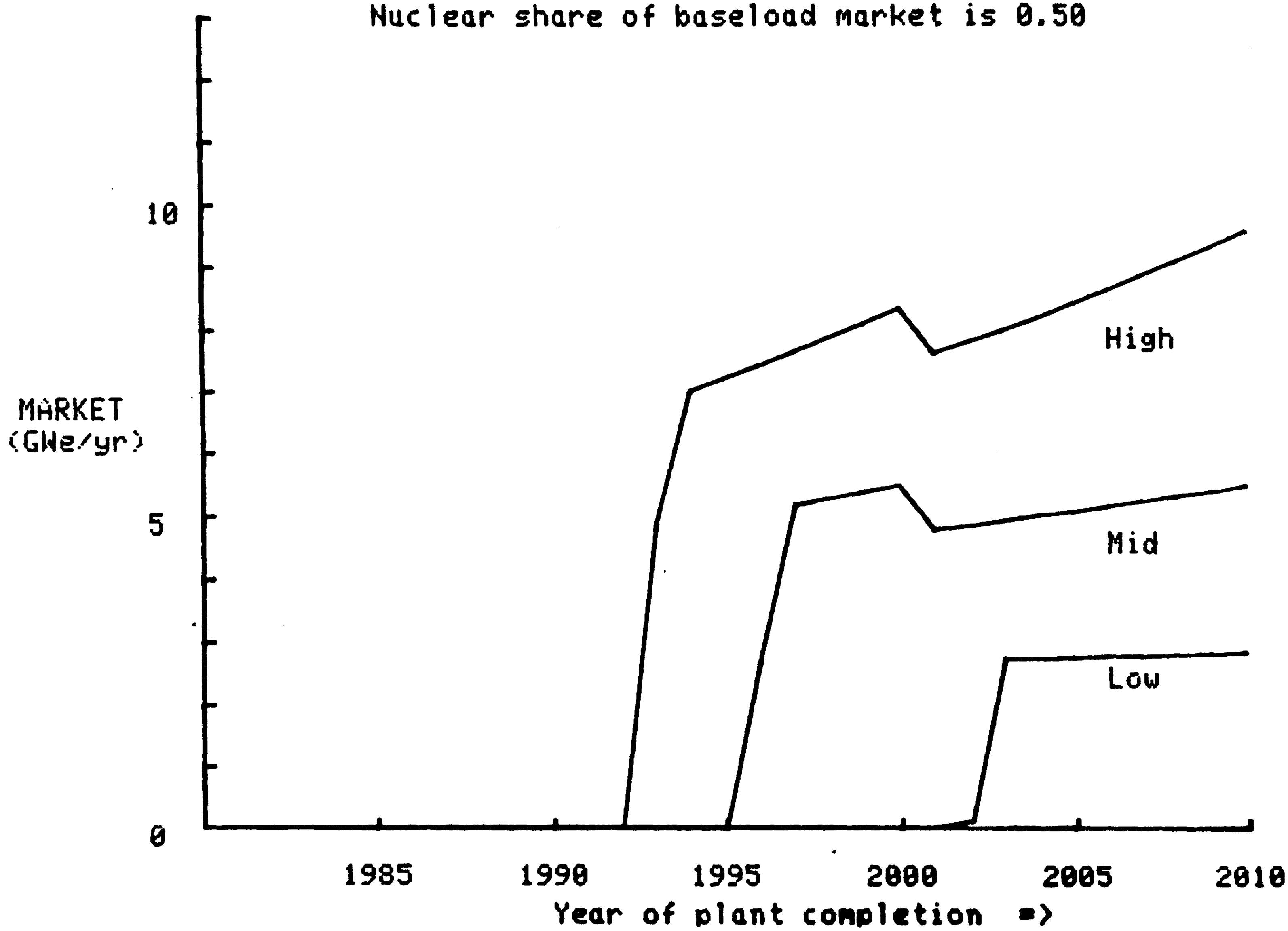


Fig. 3.4
NEW NUCLEAR PLANT MARKET: REFERENCE CASE
Nuclear share of baseload market is 0.25

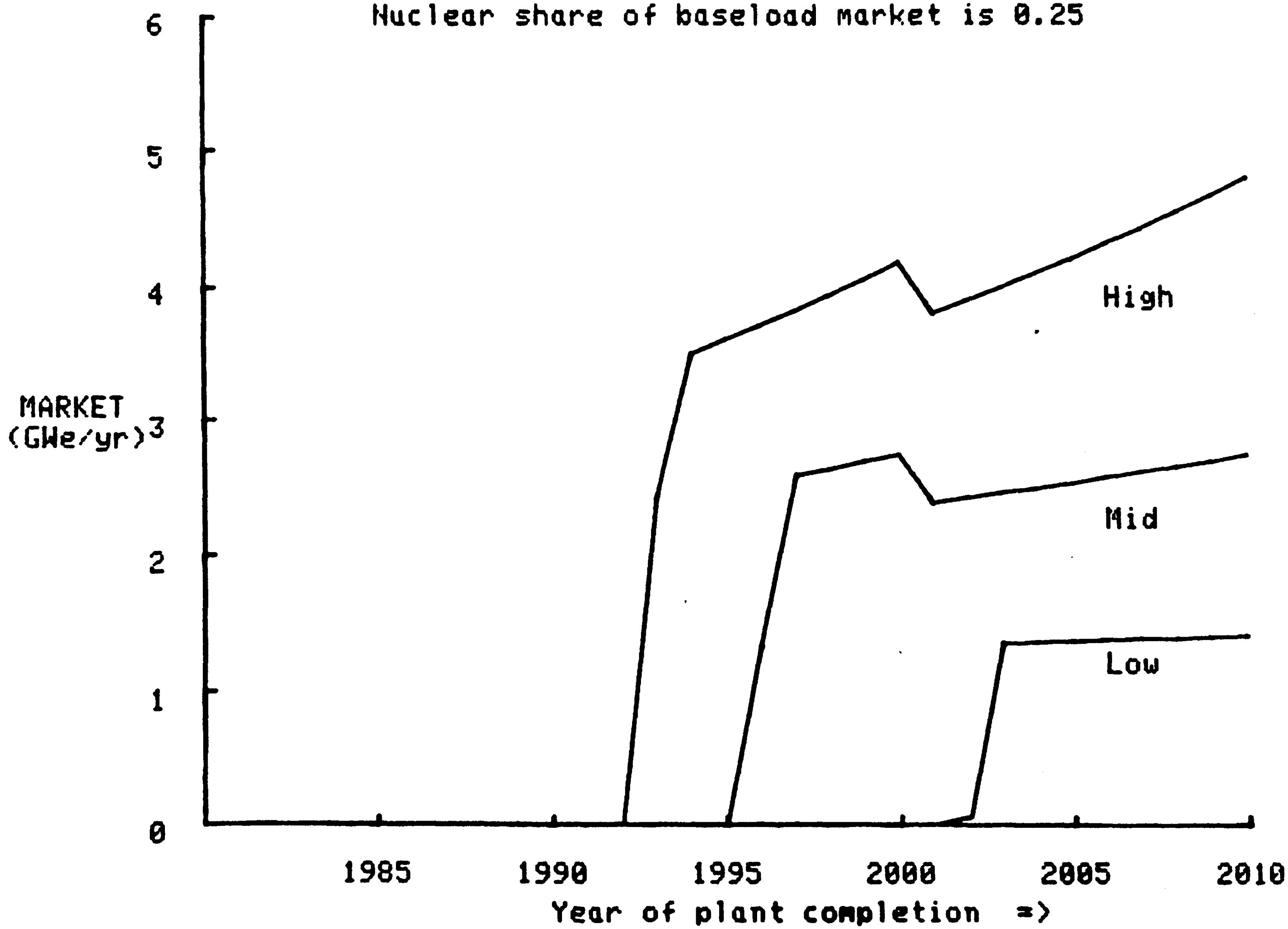
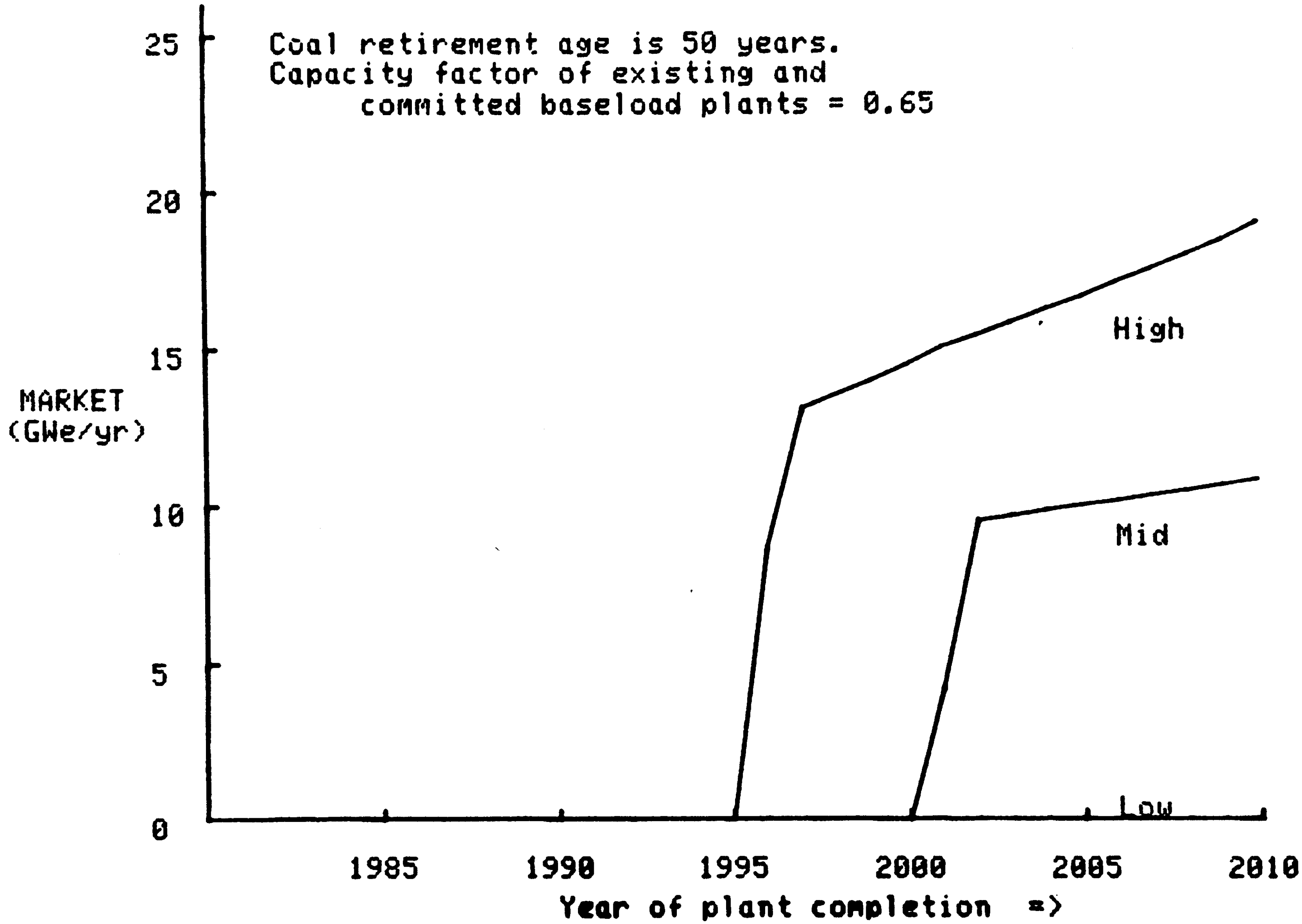


Fig. 3.5
NEED FOR NEW BASELOAD PLANTS: MAXIMUM DELAY SCHEDULE



Appendix to Section 3

Baseload plant market model

A very simple model of the U.S. electricity supply system was developed in order to predict the first year in which as yet unordered baseload power plants would be required to enter service to satisfy electricity demand, and to estimate the market for new baseload plants (in GWe/yr) for subsequent years until the year 2010.

Three electricity demand growth scenarios were selected:

Assumed Growth Rate (%/yr)

	1983-1990	1991-2000	2001-2010
HIGH	3	3.5	3
MID	2	2.5	2
LOW	1	1.5	1

The model incorporates several other assumptions:

1. Plant retirement rates and productivity losses:

Since the objective is to project the need for new baseload plants, only baseload plant retirements and productivity losses need be considered. The following assumptions were made:

Nuclear. No existing nuclear plants are retired before 2010; the average capacity factor of these plants, presently about 57%, is increased to 60%.

Hydro. Any baseload hydro capacity retired before 2010 is replaced by new hydro plants; the average capacity factor of these plants is unchanged.

Oil and gas. Baseload oil and gas generation will remain constant at the present level until 2010.

Coal. No coal plants are retired until 1990; thereafter, all coal units 40 years old or more are retired at a uniform rate. (The age distribution of existing coal-fired capacity was obtained from the 1980 Annual Inventory of Power Plants published by the Department of Energy.) The coal plants are assumed to operate with a capacity factor of 44% at the time of retirement.

2. Completion of plants already on order or under construction:

No as-yet unordered baseload plants will enter service until the last of the coal and nuclear plants currently under construction are completed. (In the nuclear case, those plants which have been indefinitely deferred are assumed not to be completed.*) The average capacity factor of these plants does not exceed 60%. Further, they will account for all of the total marginal (i.e. new and replacement) electricity demand until their capacity factor limit is reached.**

3. New baseload plants:

After full utilization of all currently committed plants has been achieved, the share of the total marginal electricity demand that will be captured by new baseload coal or nuclear power plants will be 60%. The remainder will be provided by new intermediate or peaking units, old baseload fossil units downgraded to intermediate/peaking duty, and unconventional or small-scale baseload capacity (e.g. geothermal, hydro, solar, cogeneration, etc.).*** The new baseload plants will operate at an average capacity factor of 65%.

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* Excluding the indefinitely deferred units, 54 plants (58.8GWe) are currently under construction (U.S. NRC, Nuclear Power Plants: Construction Status Report, Vol. 6, No. 2, NUREG-0030, 1982, as modified by "World List of Nuclear Power Plants," Nuclear News, February 1983). In fact, at least 6 of the 54, none of them more than 30% complete as of February 1983, are in danger of being abandoned during the next two or three years. The amount of coal capacity under construction is 76 GWe. (1981 Inventory of Power Plants (Table 2).)

** The Energy Information Administration estimates that the coal and nuclear plants entering service between 1980 and 1990 will supply about 97% of the increase in electricity consumption during this period. (1982 Annual Energy Outlook, DOE/EIA-0383(82), April 1983,p. 129.)

***The 60% share assumed here is obviously well below the share that will be taken by new baseload plants during the coming decade, (see the preceding footnote), and is probably a conservative estimate for all but the very low growth scenarios through the year 2010.

4. INNOVATION GOALS

The strength of the competition that nuclear power will face from coal in the market for new baseload plants in the 1990s is highly uncertain. Much will depend on the severity of the emission controls that are eventually adopted in response to the acid rain problem. On the other hand, by the 1990s one or more of several advanced coal technologies now under development may offer substantial economic advantages over conventional coal-fired steam cycle plants for baseload applications. Among the leading candidates are moving-bed and entrained gasification-combined cycle systems with high temperature turbines, and atmospheric and pressurized fluidized bed combustion systems. The pending environmental controls may in fact hasten the introduction of these technologies, as their emissions are usually lower and more easily controllable than those from conventional coal plants.

But even if the exact nature of the competition from coal is difficult to predict, the most important general targets for nuclear power plant innovation efforts are clear from the performance of the current generation of LWR plants. These targets are listed in Table 4.1. We have grouped them into two general categories, based on our judgement of their relative importance. Neither the list nor the ranking should be regarded as definitive, since time did not permit a systematic survey of current and prospective plant owners concerning this question. Moreover, no secondary ranking was attempted nor should be inferred within each category. Each of the first rank of goals is discussed briefly in the following paragraphs.

Table 4.1

<u>The main targets of technological innovation</u>

A. <u>First rank</u>

- Lower and more predictable plant capital costs
- Shorter and more predictable construction lead-times
- Higher reliability
- Reduced financial and health and safety risks (perceived and actual)
- Optimized plant size

B. <u>Second rank</u>

- Lower fuel cycle costs
- Lower operating and maintenance costs
- Lower occupational radiation exposures
- Higher thermal efficiency
- Enhanced load-following capability

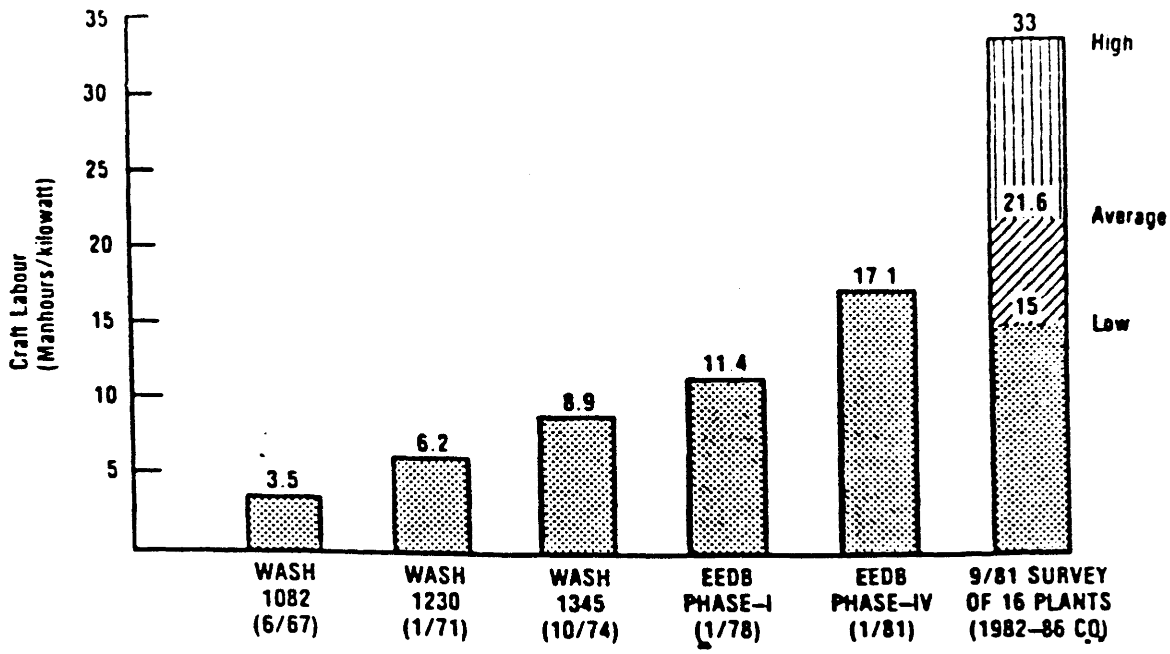
(i) Lower and more predictable plant capital costs

During the past 15 years, average capital costs of U.S. nuclear plants have escalated at a compound rate of at least 20% per year -- roughly twice the rate of inflation over this period.* The multiplicative effects of high interest and inflation rates and lengthening construction times have been a major factor underlying this trend. (In a recent DOE capital cost projection for a hypothetical nuclear plant begun in 1981 and entering service in 1993, escalation and interest during construction account for about 70% of the final cost.**) But there have also been dramatic increases in base construction costs (i.e., the costs of construction materials, site labor, engineering services, equipment and components, etc.), reflecting the greatly increased complexity of the more recent plants. Figure 4.1 shows the rapid growth in requirements for construction materials and the sharp decline in craft labor productivity experienced during the last decade.

At least as striking as the increase in average construction costs is the extraordinary variation in these costs across the country. As Fig. 4.2 shows, the most expensive plant to be completed during the coming year is more than a factor of three costlier than the cheapest. Figure 4.1 provides a glimpse of the widely varying materials and labor requirements for plants entering service during the next few years, but a systematic analysis of the factors responsible for these cost differences has yet to be carried out.

* J.H. Crowley and J.D. Griffith, "U.S. construction cost rise threatens nuclear option," Nuclear Engineering International, June 1982, 25-28.

**Crowley and Griffith, op. cit.



Trend in PWR craft labour requirements

Trends in PWR construction material requirements

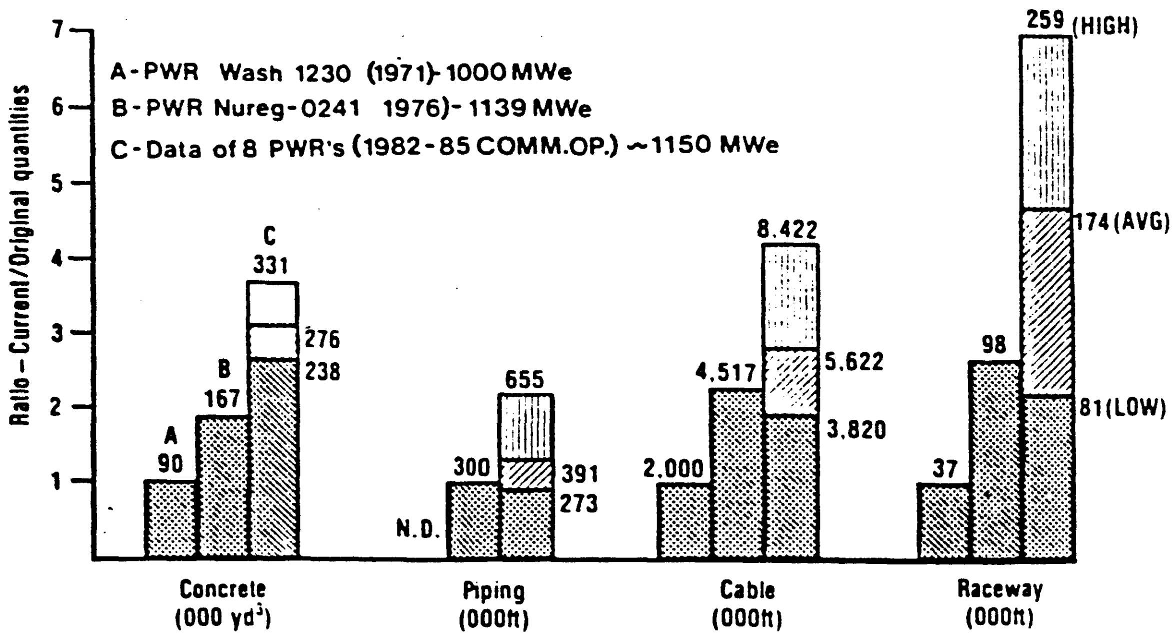
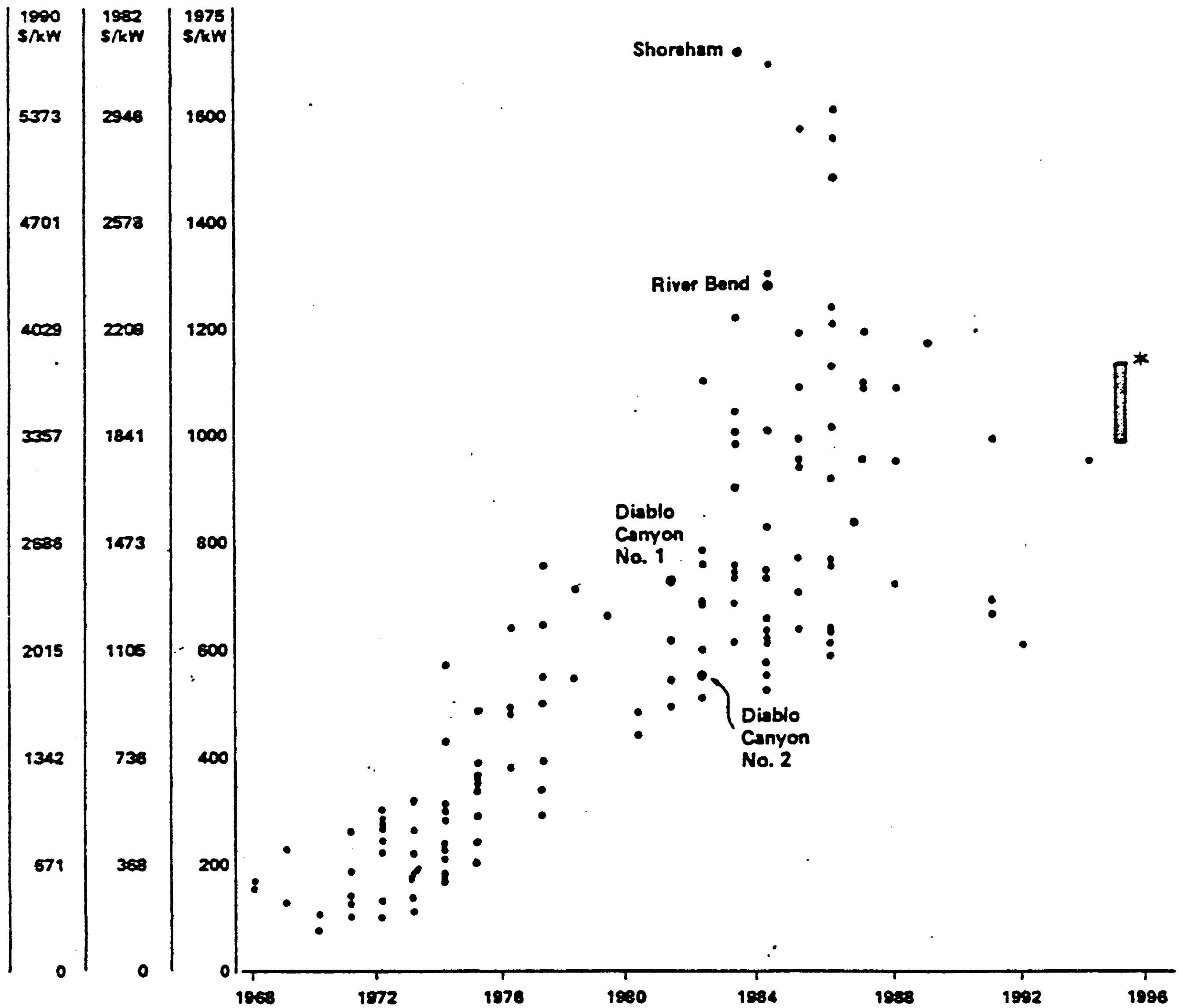


Fig. 4.1 Trends in PWR construction material and labour requirements (source: Crowley and Griffith)

Nuclear Plant Construction Costs: Cost per kW



* Range of projected costs for addtional new plants in 1995, based on Projected Cost of Electricity from Nuclear and Coal-Fired Power Plants, EIA, August 1982.

Includes all costs (both construction expenditures and accumulated AFUDC, where applicable).

SOURCE: EIA Form 254

Fig. 4.2 Nuclear plant construction cost trends.
(source: Report of the Electricity Policy Project, U.S. Department of Energy, June 1983)

Whatever the causes, there is no doubt that the most egregious examples of nuclear power plant cost overruns, and the financial troubles they have created, stand as a highly undesirable precedent and a significant deterrent to new nuclear ordering.

But there is also reason to believe that even the 'average' performance of the industry with regard to nuclear power plant construction is no longer adequate. If the nuclear option is to be restored to competitiveness, not only must aberrations of the Shoreham type be prevented, but the average plant construction cost must be reduced below the present level.

(ii) Shorter and more predictable construction lead-times

Construction lead-times for nuclear power plants in the United States have doubled during the last decade (see Fig. 4.3). In some instances, schedules have been deliberately stretched out by utilities confronting stagnating or declining demand or cash flow problems. In other cases, management and labor problems, poor quality control, and retrofit requirements arising from regulatory changes have played a role. In general, the rapidly growing complexity of power plant designs, driven by increases in unit size and by the increasing number and stringency of nuclear safety and environmental regulations, has been a key contributor to the lead-time trend.

The impact on capital costs has already been noted. An additional effect has been that lead-times now typically extend beyond utility planning horizons. The latter problem has been compounded by the increasing uncertainties associated with electricity demand forecasts. In such a situation,

CONSTRUCTION LEAD TIMES FOR NUCLEAR POWER PLANTS IN THE U.S.

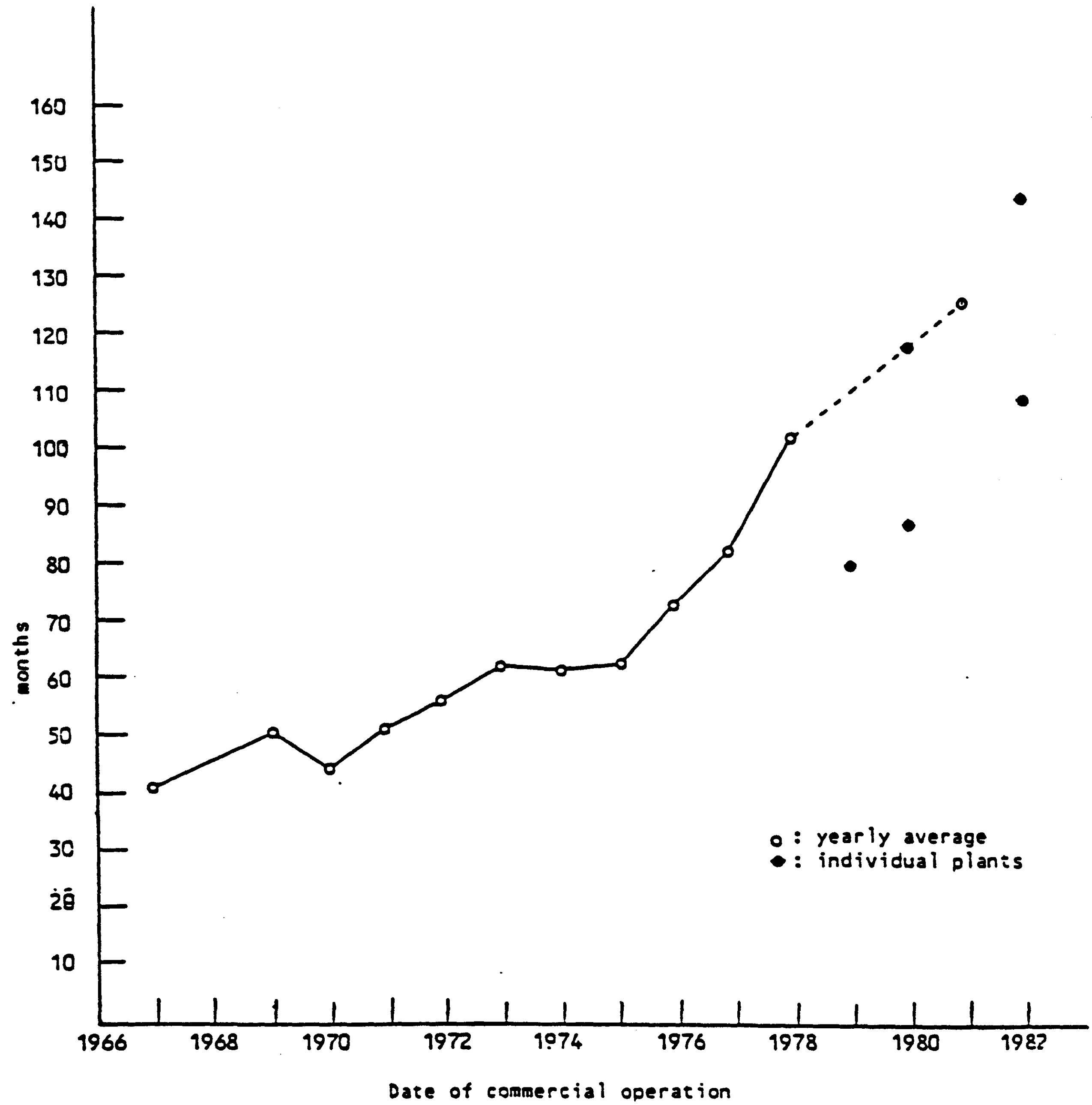


Fig. 4.3 Construction lead times for nuclear plants in the U.S.

Source: Adapted from Richard K. Lester, Nuclear Power Plant Lead Times, Report to the International Consultative Group on Nuclear Energy, Rockefeller Foundation/Royal Institute for International Affairs, New York, 1979.

planners will opt for capacity additions with shorter and more predictable lead times, and will probably be prepared to incur a generation cost penalty in return for the reduction in uncertainty so obtained. Given the volatility of the present demand outlook, even the fastest construction schedule of which the industry is currently capable -- a 72-month interval from construction permit issuance to full-power operation recorded by Florida Power and Light with St. Lucie 2 -- when combined with a typical preconstruction lead-time (i.e., the time required for site selection and approval and issuance of the construction permit) of three years or more, may still exceed the prudent planning limit for many utilities. Even with a more predictable demand outlook, a significant reduction below the recent average construction lead-time of 120 months would be necessary to restore the competitiveness of the nuclear option.

(iii) Higher plant reliabilities

The operating performance of most U.S. nuclear power plants thus far has been a disappointment. Despite early predictions that plant capacity factors would approach or even exceed 80%, only 5 of the 72 U.S. light water reactors in commercial operation as of late 1982 had achieved lifetime capacity factors of 75% or more, whereas the cumu-lative performance of almost two-thirds of the plants had failed to exceed 60%. For the last several years, the average annual capacity factor for all U.S. plants has ranged from 55 to 60%.

Several recent analyses have sought to explain the wide variation in plant performance in terms of unit size, age, design vintage, manufacturer, and other factors.* There is good evidence that reliability decreases with increasing plant size, and other evidence that it improves as plants age. (See Tables 4.2 and 4.3.) But much of the variation cannot satisfactorily be explained in such general terms. In any event, there is a strong economic incentive to improve plant performance. In today's economic conditions, an increase in capacity factor from 60% to 80% would reduce the cost of nuclear electricity by about 20%. Put differently, a recently completed light water reactor power plant operating on the once-through fuel cycle with a capacity factor of 80% could afford to burn uranium costing almost \$180/lb U_3O_8 during the forthcoming year (i.e., about seven times more expensive than today's spot market price) and still produce electricity during that period at the same overall cost as an average performer among current plants with the same capital cost.** For the more expensive plants shortly to be completed, an even higher uranium cost could be offset by the same improvement in capacity factor.

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* See, for example, S. Thomas, "Worldwide Nuclear Plant Performance Revisited," Science Policy Research Unit Occasional Paper Series No. 18, University of Sussex, 1982; C. Komanoff, Power Plant Performance -- Nuclear and Coal Capacity Factors and Economics, Report No. 56-1, Council on Economic Priorities, New York 1976; P.L. Joskow and G.A. Rozanski, "The effects of learning by doing on nuclear plant operating reliability," Rev. Econ. Stat., Vol. LXI, May 1979, No. 2.

**For this comparison, the average plant was assumed to have a capacity factor of 60%. The capital cost of both plants was assumed to be \$1500/kw, and a capital charge rate of 20% was used. The discharge fuel burnup was assumed to be 30,000 MWD/MT, and the plant thermal efficiency was 32%.

Table 4.2

U.S. Reactor Performance: Lifetime Load Factors*

	500-700 MWe No. of Reactors	%	700-1000MWe No. of Reactors	%	1000+Mwe No. of Reactors	%
Westinghouse	6	77.2	8	55.0	8	57.6
General Electric	7	64.0	9	55.6	5	65.0
Babcock and Wilcox			9	47.6		
Combustion Eng.			6	66.6		

* Mean lifetime load factors for each reactor exclude years 1 and 2 of operation.

Source: S. Thomas (1982)

Maturation Profile of US Reactors

% Load Factor (No. of Reactors)

Year of Operation	500-700 MW*		700-1000 MW				1000+ MW	
	Westinghouse	GE	Westinghouse	GE	Combustion Eng	B & W	Westinghouse	GE
1	64.1 (6)	55.4 (7)	59.4 (8)	58.6 (10)	61.3 (7)	48.9 (9)†	49.7 (8)	40.7 (5)≠
2	68.9 (6)	53.4 (7)	55.2 (8)	55.1 (10)	53.4 (6)	51.7 (9)†	54.0 (8)	53.1 (5)≠
3	77.9 (6)	58.6 (7)	57.9 (8)	54.3 (9)	63.0 (6)	50.3 (8)†	51.8 (8)	56.6 (5)
4	77.2 (6)	61.4 (7)	52.6 (7)	50.2 (9)	64.0 (6)	66.2 (8)	57.7 (7)	71.7 (5)
5	79.5 (6)	65.2 (7)	71.1 (6)	53.9 (8)	68.4 (5)	50.0 (6)†	57.2 (6)	74.9 (4)
6	72.4 (6)	66.2 (7)	56.5 (5)	60.2 (7)	69.0 (4)	44.8 (6)†	62.5 (5)	67.5 (4)
7	76.4 (6)	77.3 (6)	48.1 (5)	63.2 (5)	56.3 (2)	54.9 (5)†	59.9 (3)	51.0 (3)
8	82.7 (4)	65.7 (6)	64.6 (5)	55.9 (4)	48.3 (2)	39.8 (1)	52.0 (2)	
9	75.7 (4)	69.4 (6)	33.2 (3)	67.0 (4)	62.0 (2)			
10	65.9 (2)	62.2 (4)	57.7 (1)	62.2 (2)				
11	70.3 (2)	60.3 (3)						
12		54.3 (2)						

Notes

* The one reactor supplied by Combustion Engineering in this size range (Fort Calhoun) is excluded, since they have no orders for this size of reactor and evidence on one reactor is of limited value. Generally its performance has been good.

† Years 1, 2 and 3 of the Babcock & Wilcox data are affected by Three Mile Island 2 which produced virtually no power in that period. Similarly years 5, 6 and 7 are affected by the shutdown of Three Mile Island 1 following the same accident.

≠ More than 70% of the first year of operation of two of the Browns Ferry units was lost due to the cabl tray fire, and more than 70% of the second year of operation of unit one was lost for the same reason.

Table 4.3 Maturation profile of U.S. reactors (source: S. Thomas (1982))

(iv) Risk reduction

Since the accident at Three Mile Island, discussions of nuclear power plant risks have emphasized the need to differentiate between the risks to public health and safety and the environment and the financial risks to the plant owners. Probabilistic risk analyses suggest that on neither count are nuclear plants on average any worse than other large-scale energy facilities which appear to be generally accepted.* Thus, is there a rational case for attempting to reduce the risks from nuclear power plants any further?

For the prospective innovator, the pragmatic answer is that a case surely can be made if the product would be acceptable where current plants are not. But this in turn introduces the difficult question of the relationship between perceived and actual risk. In exploring this issue, several authors have proposed risk definitions broader than the simple measure of expected mortality rate (i.e., the product of event probability and number of fatalities per event) in an attempt to match more closely the risk perceptions of the lay public.** These studies indicate that public risk

* The literature on this subject is extensive. For a recent discussion, see A.M. Weinberg, "A Second Nuclear Era: Prospects and Perspectives," presented at the 40th Anniversary Celebration of the First Nuclear Chain Reaction, University of Chicago, December 1-2, 1982.

**See, for example, Rowe, W.D., An Anatomy of Risk, (New York, Wiley 1977); D. Litai, D.D. Lanning and N.C. Rasmussen, "The public perception of risk," Proceedings of the Society of Risk Analysts, June 1981 Meeting, Washington, D.C.; and C. Hohenemser, R.W. Kates and P. Slovic, "The nature of technological hazard," Science, Vol. 220, 22 April 1983, 378-384.

perceptions vary non-linearly with the magnitude of the event consequence, implying that more could be gained in terms of enhancing the acceptability of nuclear power plants by reducing the publicly perceived probability of catastrophic events than the traditional, one-dimensional measure of risk would suggest. Several other factors have also been identified which lead people to feel differently about hazards for which the products of frequency and consequence are the same. One goal of current research is to determine the strength of societal risk aversions to catastrophic events and to other characteristics of hazards, and to examine the variation in the strength of these risk aversions among different social groups.* Such information is important in determining the appropriate targets for engineering responses to public concern about technological hazards.

A related issue concerns the possibility that different technological approaches to the same target might elicit different public reactions. For example, a technical response to the societal risk aversion to catastrophic events which relied on a collection of engineering modifications to existing designs whose effectiveness could only be discerned through the application of probabilistic risk assessment methods might be less convincing, by virtue of its greater obscurity to a lay audience, than one which was based on more easily understood or demonstrable physical principles, even if the actual reduction in the probability of a catastrophic accident was the same in both cases. (The presumed virtue of 'transparency' in reactor safety philosophy was an important factor in the development of the PIUS concept (see

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* Litai et al, op. cit.

Section 6.2).) Nevertheless, our ability to gauge accurately the practical significance of such differences is discouragingly low.

More generally, although the strategy of devising engineering 'fixes' to shape public perceptions might initially seem attractive, the volatility of these perceptions, their susceptibility to uncontrollable external influences, and the generally poor understanding of how they develop and how they affect public policy may mean that in practice such a strategy is unworkable.

A less ambitious, but potentially more rewarding approach would be to concentrate on the risk perceptions of utility managements. As is often pointed out, they, not the general public, make the decisions to order nuclear plants. But this approach is also not without difficulty. For example, the preoccupation of utility owners and managers with the financial risks of nuclear investments is not really separable from public perceptions of health risks. The latter bear directly on the former, for example via political and regulatory actions which may complicate and prolong construction projects, or which may result in expensive shutdowns even in the absence of any physical damage to the plant. Moreover, as with public perceptions of health risks, it seems plausible that plant owners' perceptions of financial risks are shaped by more than probabilistically derived expected dollar losses. Once again, our present understanding of this question is not well enough developed to point reliably to a sensible design strategy, though the problems of measurement and analysis seem more tractable in this case, because of the much smaller and less diverse population at issue. One feature of utility risk perceptions already seems clear, however: no risk-reducing design innovation -- whether it involves an evolutionary change in current LWR designs or a radical departure from them -- is likely to capture the interest of U.S. utilities unless it is accompanied by an overall reduction in the expected cost of nuclear electricity generation.

(v) Optimized unit size

There has been much inconclusive discussion in recent years of whether the economies of scale putatively realized in large nuclear plants would be more than offset by the higher reliabilities, inherently lower risks, less complex designs, and shorter construction times expected of future smaller units (and actually experienced with smaller units in the past). The discussion has been handicapped by a lack of practical experience with small plant design and construction in the present economic and regulatory environment. The question of appropriate unit size must also be addressed from the perspective of utility system capacity needs and financial planning, however, and here the situation is clearer. Given the financial constraints and lower load growth behavior which seem likely to prevail for the foreseeable future, we anticipate that most utilities, including some of the largest, will opt to build new capacity in smaller increments than the 900-1200 MWe range typical of current nuclear plant designs. Moreover, in a period of increasing marginal capital costs, smaller units can generally be introduced into a utility's rate base with much less public resistance than the large plants. We note in passing that the trend of the advanced coal technologies mentioned previously is also towards smaller unit sizes.

To investigate the unit size issue further, a simple, highly stylized model of the U.S. electric utility industry was constructed. The purpose of the model was to estimate the impact of unit size on the theoretical upper limit of nuclear penetration into the new baseload plant market during the period 1990-2010. The model and results are summarized in the Appendix to this Section. Among the results: the potential expansion of the nuclear market to include utilities too small to accommodate large nuclear units which would result from reductions in unit size seems unlikely to be a decisive factor in determining whether such reductions would be worthwhile. Rather, the issue would seem to hinge on whether the previously discussed advantages claimed for the smaller units are in practice sufficient to attract the interest even of those utilities whose systems are large enough to accommodate the large plants.

Fuel cycle costs

In table 4.1, the goal of reducing fuel cycle costs was included in the second rank of innovation targets. This ranking differs from the traditional view of the role of nuclear power plant innovation. The latter, formulated during a period of rapid expansion of the nuclear industry, was based on the expectation that depletion-driven increases in the price of natural uranium would be the primary threat to the economic competitiveness of the nuclear option. According to this view, the chief goal of reactor design innovation was to ensure that nuclear power plant systems with suitably conservative uranium consumption characteristics would be available commercially on a schedule dictated by the rate of uranium price escalation The efforts in the U.S. and overseas to develop the liquid-metal cooled fast breeder reactor (LMFBR) are, of course, the product of this thinking.

As detailed above, however, other, more immediate threats to the competitiveness of the nuclear option have emerged within the last decade which are at least partly susceptible to mitigation through technological innovation, but which the breeder was not intended to address. At the same time, as a result of both sharp reductions in the expected growth rate of nuclear power and recent additional discoveries of high-grade, low-cost uranium resources, the projected future rate of increase of uranium prices has declined, and the need to develop an appropriate technical response has become correspondingly less urgent.

There may well be a time in the future when high uranium costs will induce a shift to the LMFBR or to an alternative breeder concept. But those circumstances now seem very unlikely to arise before the year 2025 in the U.S., and certainly will not arise at all if the other problems which are currently inhibiting nuclear power growth are not resolved first.

Conceivably, innovations introduced to resolve these nearer-term problems could become a victim of their own commercial success; that is, they could be driven into premature obsolescence by depletion-driven uranium price increases and resulting competition from the breeder before their development costs could be fully recovered. We did not attempt a quantitative analysis of this issue in the present study; the answer obviously would depend on the magnitude of the initial investment required to commercialize the innovation, as well as on trends in electricity demand and uranium supply well beyond the 2010 cut-off date used in the analysis in Section 3. The issue certainly deserves more careful study. However, even without such an assessment it seems probable that, in decisions on whether to make major new investments in nearer-term nuclear power plant design innovations, the financial risk of premature obsolescence posed by the breeder would be outweighed by the risk that the product would fail to attract sufficient utility interest in the first place.

Appendix to Section 4

Impact of nuclear unit size on ordering potential

To investigate the issue of appropriate unit size in more detail, a simple, highly stylized model of the U.S. electric utility industry was constructed. The model disaggregates the national utility system into individual utility service areas. The purpose of the model was to estimate the impact of unit size on the theoretical upper limit of nuclear penetration into the new baseload plant market during the period 1990-2010. In other words, by how much would a given reduction in nuclear plant size potentially increase the number of nuclear orders placed during this period? The model was based on the following principal assumptions:

- In 1990, and again in the year 2000, each utility would order one or more new baseload plants based on its 10-year forecast of demand growth in its service area and its expected 10-year schedule of plant retirements.
- Each utility would order the combination of units which would approach as closely as possible its baseload capacity requirements at the end of the tenth year without exceeding them. (A further, implicit assumption is that baseload plant lead-times will not exceed 10 years.)
- There is no joint utility ownership of plants.

Obviously this last assumption will bias the model in favor of smaller units. In practice, co-ownership is already quite common in the utility industry, and is probably increasing in extent.

Three national electricity demand growth scenarios were selected corresponding to those analyzed in Section 3, as shown below:

Assumed Growth Rate (%/yr)

	1990-2000	2000-2010
LOW	1.5	1.0
MID	2.5	2.0
HIGH	3.5	3.0

For each national scenario, average growth rates were obtained for each of the nine electric reliability regions in the U.S. These derived regional growth rates reflect the current pattern of interregional differences in electricity demand. All utilities in each region were assumed to experience the average regional growth rate. (A fuller description of the assumptions used in the model is given at the end of this Appendix.)

In order to estimate the theoretical upper limit on nuclear ordering, it was assumed that a nuclear unit would always be preferred to a coal unit of the same size. Utility ordering behavior was then simulated for each of the three demand scenarios assuming that the smallest nuclear unit available was (1) 1000 MWe; (2) 600 MWe; (3) 300 MWe; or (4) 100 MWe.

Figure 4.A.1 shows, for the 'MID' scenario, the maximum share of marginal 'baseload' demand between 1990 and 2000 that could be captured by nuclear for the four unit size cases. Also shown is the maximum nuclear market share if nuclear ordering were limited to those utilities which already had nuclear plants in operation prior to 1990. Figure 4.A.2 shows the results for the decade 2000-2010. Figures 4.A.3 and 4.A.4 show the number of utilities placing orders in each case.

The main points emerging from this analysis are as follows:

- The utilities that are individually large enough to introduce at least one 1000 MWe unit in each of the 10-year planning cycles will account for almost half of the industry's total projected requirement for new baseload capacity.
- More than 70% of these large utilities will have had prior nuclear experience by 1990. On the other hand, less than half of all the utilities with prior nuclear experience by 1990 will be in a position to order new 1000 MWe units before 2010.
- The availability of smaller units increases the potential market penetration of nuclear significantly, though the actual effect would be smaller than is indicated in the figures because co-ownership would result in a greater commitment to the large plants. With no co-ownership, the availability of 300 MWe units would increase the

Fig. 4.A.1

PERCENTAGE OF MARGINAL BASELOAD DEMAND POTENTIALLY SATISFIED BY NUCLEAR UNITS: 1990-2000 (MID-CASE)

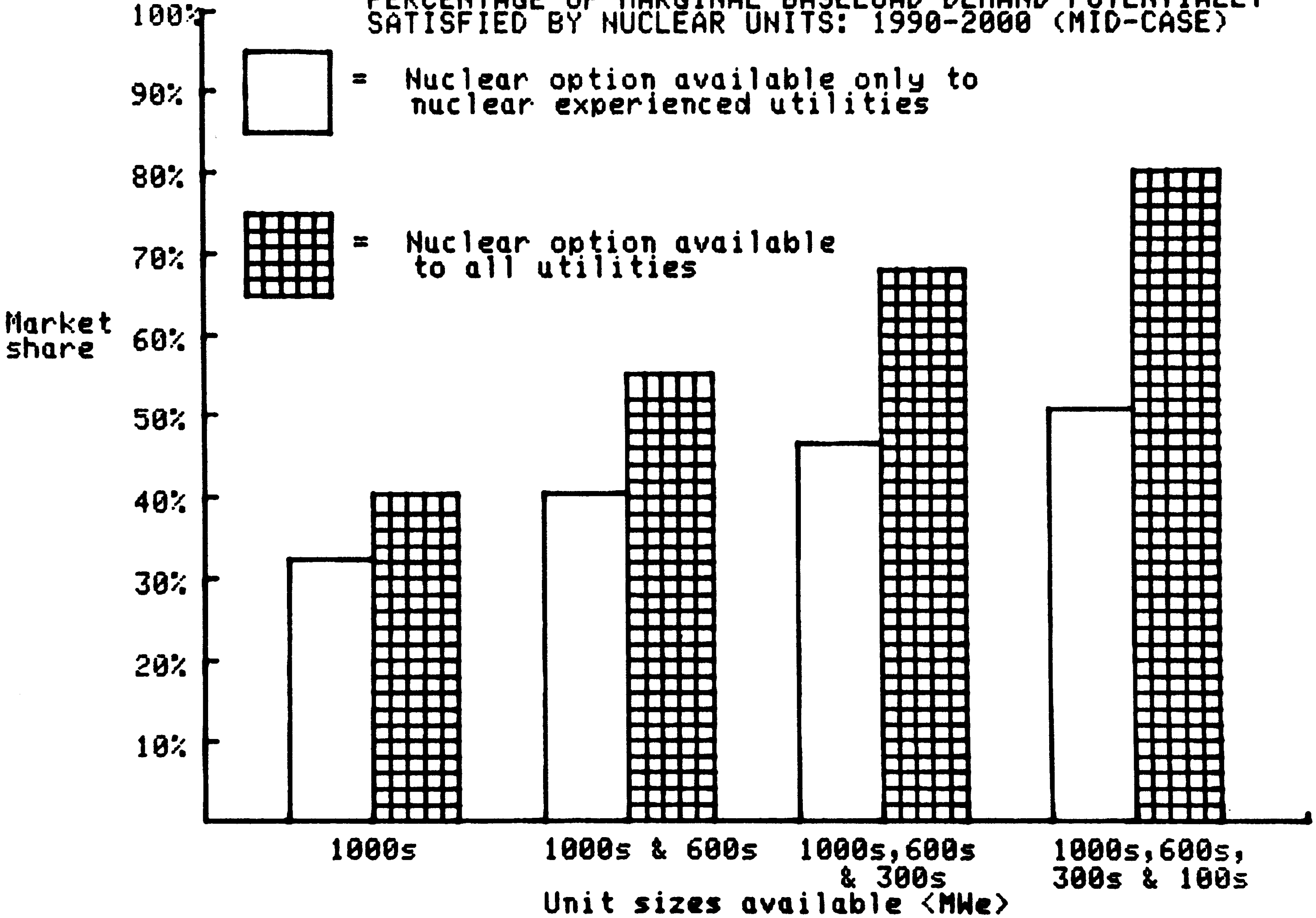


Fig. 4.A.2

PERCENTAGE OF MARGINAL BASELOAD DEMAND POTENTIALLY SATISFIED BY NUCLEAR UNITS: 2000-2010 (MID-CASE)

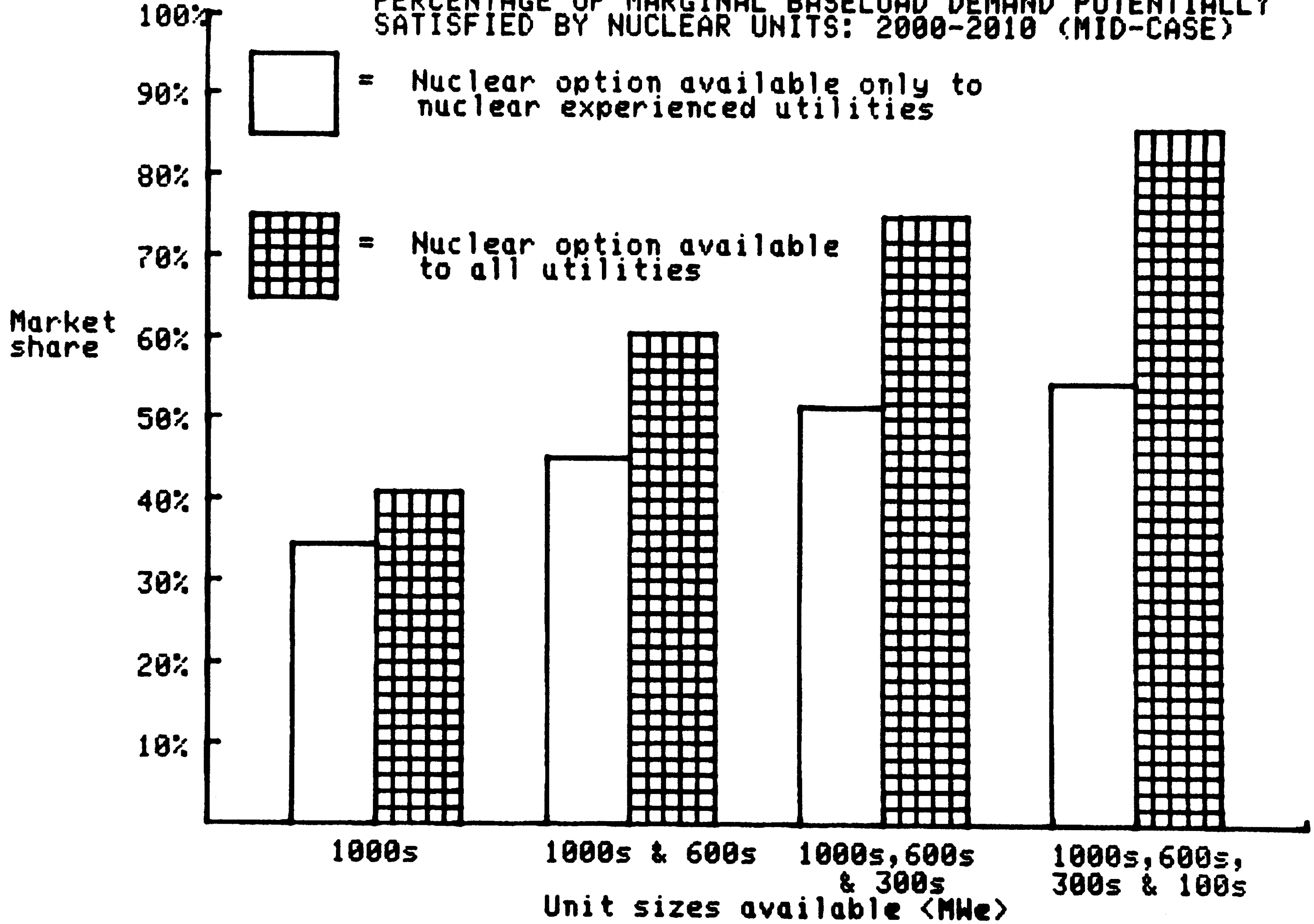


Fig. 4.A.3

NUMBER OF UTILITIES CAPABLE OF PURCHASING NUCLEAR UNITS: 1990-2000 (MID-CASE)

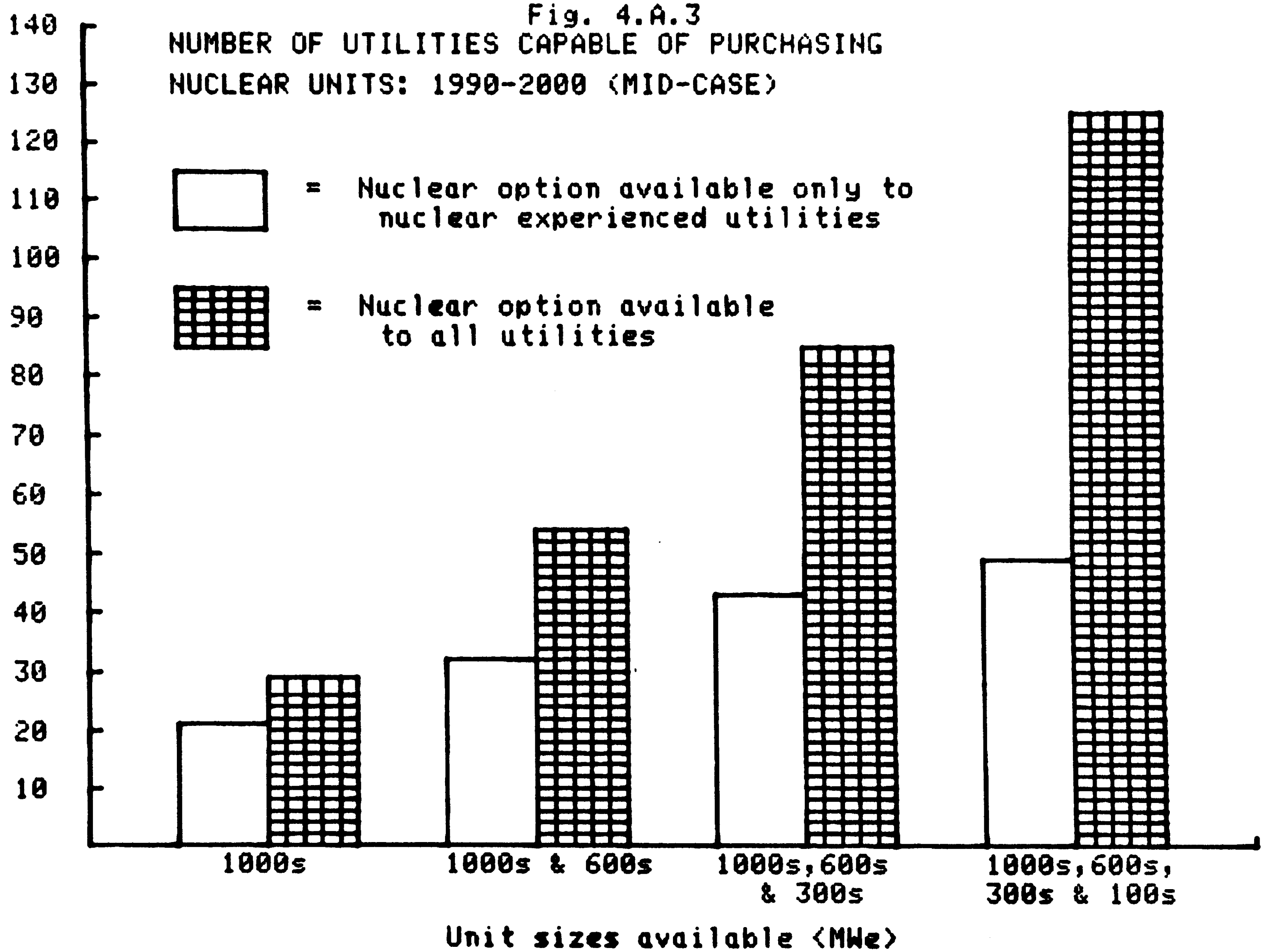
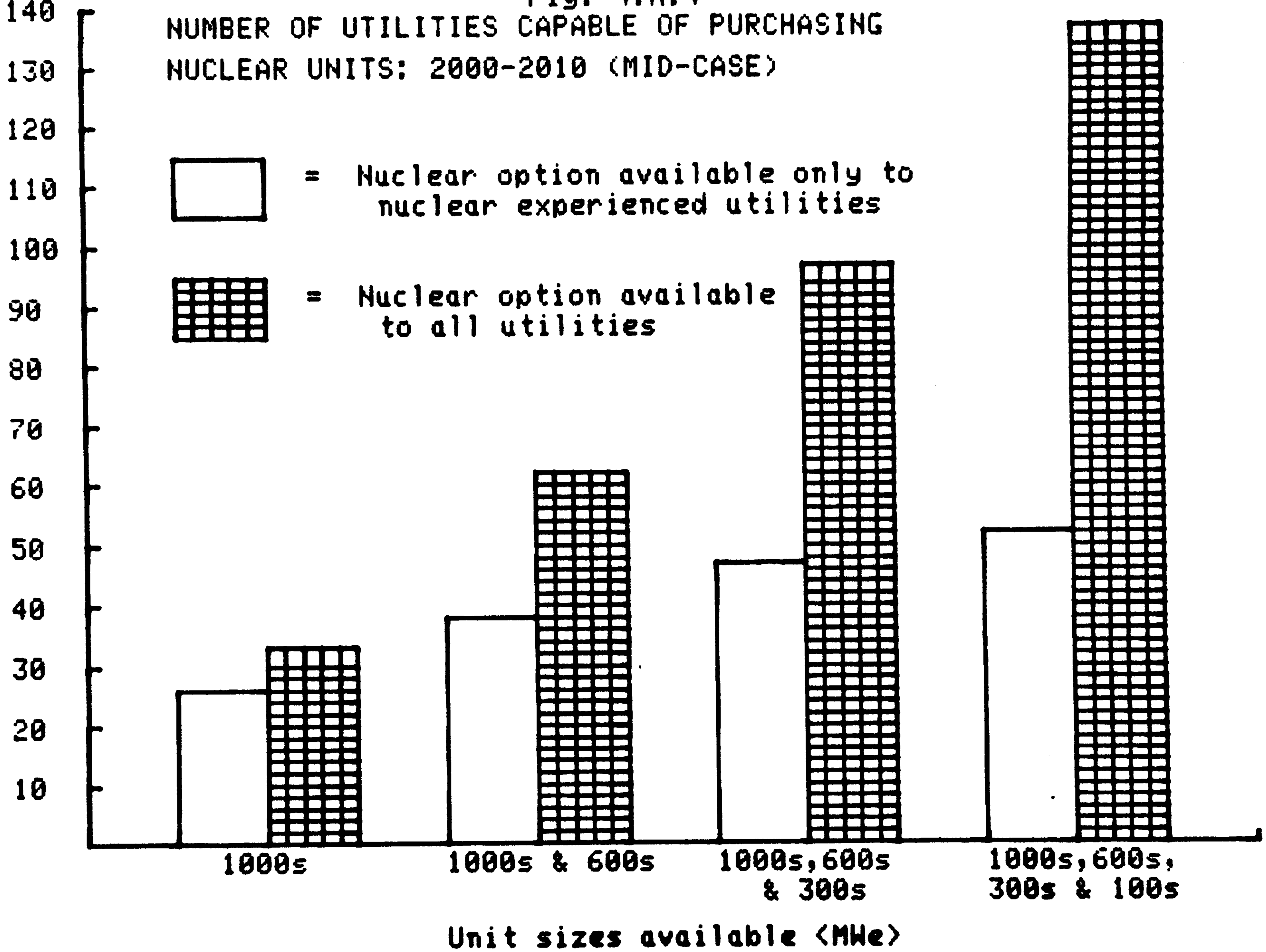


Fig. 4.A.4


NUMBER OF UTILITIES CAPABLE OF PURCHASING
NUCLEAR UNITS: 2000-2010 (MID-CASE)
140
130
120
110
100
90
80
70
60
50
40
30
20
10
= Nuclear option available only to nuclear experienced utilities
= Nuclear option available to all utilities
1000s
1000s & 600s
1000s, 600s & 300s
1000s, 600s, 300s & 100s
Unit sizes available (MWe)

maximum nuclear market share by 70% relative to the 1000 MWe case during the 1990s (equivalent to an increase in the maximum potential ordering rate of about 3 GWe/yr.)

- If the increase in the potential size of the nuclear market created by the availability of smaller units were to be exploited fully, a majority of the utilities ordering the smaller plants would have had no previous nuclear experience. Ease of construction and operation would thus be particularly important design goals for small systems.
- Though a more detailed analysis of the issue is certainly necessary, a plausible interpretation of these preliminary results is that the expansion of the potential market to include utilities which could not accommodate large units on their system is unlikely to be a decisive factor in determining whether significant nuclear unit size reductions would be worthwhile. Rather, the issue would seem to rest on whether the other advantages claimed for smaller units -- discussed elsewhere in this report -- are in fact sufficient to attract the interest of those utilities which could in principle also accommodate large plants.

Other Major Assumptions

(1) Regional growth rates were formulated so that the assumed national average growth rate was maintained during each ten-year period, while the growth rate distribution from region to region reflected

that predicted by the National Electric Reliability Council for the period 1980 to 1990.*

(2) Sixty-five percent (65%) of the marginal (i.e., new and replacement) electricity supply will be generated by units on baseload duty.

(3) The average capacity factor for each utility's total generating plant was taken to be 44%,** and the capacity factor of base-loaded units was assumed to be 65%.

(4) Coal stations were retired after a 40 year service lifetime. Their capacity factor at the time of retirement was assumed to be 44%.

(5) Generation data for the approximately 350 individual utilities treated in this analysis were obtained from "Statistics of Publicly Owned Electric Utilities...1980"*** and "Statistics of Privately Owned Electric Utilities...1980."****

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* National Electric Reliability Council, "Electric Power Supply and Demand, 1981-1980," July 1981.

** U.S. Department of Energy, July 1981, "1982 Annual Energy Outlook," DOE/EIA-0383(82), April 1983.

*** U.S. Department of Energy, "Statistics of Publicly Owned Electric Utilities in the United States -- 1980," DOE/EIA-0172(8).

**** U.S. Department of Energy, "Statistics of Privately Owned Electric Utilities in the United States -- 1980 Annual (Classes A & B Companies)," DOE/EIA-0044(80).

5. TECHNOLOGICAL INNOVATION VERSUS POLITICAL AND INSTITUTIONAL REFORM

Many of the targets for technological innovation discussed in the preceding paragraphs can in principle also be achieved by institutional reforms of various kinds, some of which are already underway. For example, although its prospective contribution is sometimes exaggerated, stabilization of the nuclear power plant licensing process through administrative and possibly legislative reform can help to reduce utility concerns over the financial risks of new nuclear investments. Higher levels of insurance coverage for property damage and replacement costs incurred during forced outages can also help in this regard. Standardization of existing plant designs would facilitate the licensing process and enhance the predictability of construction costs and leadtimes. Improvements in construction management practices would alleviate many of the problems now being experienced at nuclear construction sites around the country. Better training programs for operating and maintenance personnel would help upgrade plant operating performance and reduce the risk of accidents initiated or aggravated by human error. A more sympathetic economic regulatory environment -- in particular, one in which all utilties were permitted to earn a return on construction work in progress -- would help to lessen the current utility industry aversion to long lead-time, capital-intensive nuclear construction projects.

Beyond these measures, other more far-reaching organizational reforms have also been proposed, though not so far acted on. Such reforms are intended to address the generic problems caused both by excessive fragmentation in the U.S. utility industry and by the often unwieldy tripartite approach to design and construction management involving utilities, NSSS vendors, and architect engineers.*

Evidence from abroad that nuclear power programs utilizing essentially the same technology are manageable when some or all of these conditions are present is often used in support of the case for such reforms in the United States.

Some have gone further, arguing that properly conceived and implemented institutional reforms, coupled with demonstrable progress in nuclear waste management and disposal, and barring any major mishap involving existing nuclear plants, will be sufficient to restore the viability of the nuclear option. This is all the more likely, it is suggested, if concerns about acid rain and CO_2 emissions from fossil power plants continue to escalate. Therefore, it is argued, technological innovations beyond the relatively modest efforts now underway are unnecessary. This assertion may prove to be correct. It is, however, impossible to prove in advance. And should it turn

* The proposed reforms include greatly expanded roles for nuclear service companies, and the creation of government or privately-owned regional nuclear generating consortia separate from the organizations responsible for transmission, distribution, and other generating operations. (For a recent review of such proposals, see J. Barkenbus, "An assessment of institutional alternatives for nuclear power generation," (mimeo) Institute for Energy Analysis, Oak Ridge, Tennessee, February 1983.)

out to be wrong, unambiguous evidence to that effect would come too late to permit a timely technological response, in view of the long development lead-times involved. Even if it is correct -- in the sense that utilities will at some future time resume ordering LWRs of conventional design -- that is not a reason not to proceed with the development of advanced nuclear power plant systems. In short, irrespective of whether current nuclear plant designs are in some sense 'adequate,' we see no logical basis for the notion of institutional reform as a viable alternative to technological innovation, but rather view them as being properly pursued in parallel.

6. THE TECHNICAL OPTIONS

In this chapter, opportunities for design innovation involving several nuclear power plant systems are assessed in light of the innovation targets established previously. Evolutionary improvements in LWR systems are examined in Section 6.1. Section 6.2 presents a discussion of the PIUS reactor concept, of Swedish origin. Current and advanced high-temperature gas reactor systems are assessed in Sections 6.3 and 6.4. The latter focuses in particular on small-scale, modular HTGR systems. Finally, liquid metal cooled breeders, heavy water moderated reactors, and several other reactor systems are reviewed in Section 6.5

6.1 Next Generation LWRs

In identifying potentially promising areas of design innovation for the next generation of LWRs, it is recognized that ultimately the value of a power station to a utility company is measured by its contribution to the financial health of the company. The financial health of a regulated utility and the economic benefits which it makes available to its customers are closely linked quantities, being roughly coincident (at least in the long run) in the areas of capital and operating costs, and in the related area of system availability. The following discussion focuses upon these areas of economic performance. The other major cost component, the fuel cycle cost, is largely external to a utility's control and to its benefit stream. Moreover, largely for historical reasons, the current generation of LWRs has been designed with greater emphasis on fuel cost minimization than on the reduction of capital and operating and maintenance costs.

The capital cost contribution to the total cost of nuclear electricity is a function of both the final construction cost of the plant and the plant capacity factor. Design innovations can affect both of these properties, and it is useful to distinguish between them in this regard.

Plant construction costs:

The main types of design innovation which influence plant construction cost are summarized in Table 6.1.1 according to the category in which benefits may be realized (e.g. carrying charges, material costs). Also shown are several project management innovations which could bring about construction cost reductions. Because of the large role played in power station design by safety requirements the safety-related portion of Table 6.1.1 is elaborated in Table 6.1.2.

Typically there may be several means of achieving the design goals summarized in these tables. For example, the goal of slowing the transient response of the PWR nuclear steam supply system could in principle be met variously by increasing the water inventories of the primary and/or secondary portions of the steam generator, by increasing the pressurizer volume, and by optimizing the moderator coefficient of reactivity. Identifying the optimal design solution in this case would require, <u>inter alia</u>, a careful quantitative specification of the design goal, along with a determination of the feasibility of the possible design options.

Plant capacity factor:

Table 6.1.3 lists the major causes of plant capacity factor reductions, and Table 6.1.4 summarizes many of the most important design measures for

Table 6.1.1

Major Factors Affecting Plant Construction Costs

Cost Component Design or Attributes	Carrying charges (Construction Duration)	Materials and Components	Labor	Satisfaction of Safety Requirements	Negotiation of Licensing Maze
Design standardization	x	x	x	x	x
Use of modular components	x	x	x	x	x
System & component simplification	x	x	x	x	x
Design for ease in construction access	x		x		
Use more smaller units at a single site	x	x	x	x	
Improvement in Safety (see Table 6.2.1	x	x	x	x	x
Project Management Attributes					
Replacement of Architect/Engineers by utility staff	?	?	?	x	?
Use of small scale design models	x		x		
Comprehensive project scheduling and management	x	x	x	x	x
Completion of design prior to start of construction	x	x	x	x	x

Table 6.1.2

Design Targets for Improving Plant Safety

Accident Prevention

Greater system reliability

Greater system simplification

Greater use of PRA to identify, upgrade and monitor sensitive components

Rendering system more forgiving of equipment failures and operator errors

- Increasing design margins
- Increasing interacting component characteristic time scales and ensuring that such scales for entire system are mutually consistent

Increasing standardization and modularization

Improving information system control scope, comprehensiveness, ability to focus upon high-priority information and reliability

Challenging the justification of systems required to deal with low risk and unrealistically stylized accidents

Designing control room for easy access and focus upon essential information

Eliminating opportunities for containment bypass in accidents

Accident Mitigation

Minimizing potential for interactions of subsystems and components which can function independently

Re-examining seismic analysis assumptions, methods and design requirements *ab initio* in light of past earthquake experience

Designing primary and containment systems to maximize fission product retention in severe damage accidents

Emphasizing use of passive heat sinks and heat removal systems

Table 6.1.3

Sources of Capacity Factor Reductions

	Capacity Factor Loss (%)*	Relative Percentage of Loss
Forced shutdown	18.4	42.1
Accident affecting plant safety		
Mechancial		
Human		
Component failure or malfunction		
Human failure or malfunction		
Unstable system response to an initiating event		
Mechanical		
Human		
Major accident	<2	<4.8
Mechancial		
Human		
Shutdown or de-rating mandated by regulatory authorities	4.7	11.3
Common mode problem		
Design error		
Reduced or lost design margin		
Maintenance shutdowns	13.4	32.1
Component repair		
Component replacement		
Radiation-limited operation		
Refueling		
Other (e.g., Training, operation below capacity)	3.2	7.7

*Source: D.C. Bley, "Light Water Reactor Productivity Improvement," MIT Nuclear Engineering Department Ph.D. thesis (1979).

Table 6.1.4

Methods for Increasing Plant Capacity Factor

Avoidance of forced shutdowns

Use of historical experience and PRA to identify unreliable components and systems to be upgraded and/or monitored

Simplification, standardization and modularization of systems

Use of more forgiving systems -- in terms of transient response time scales and of inherent design margin

Use of more reliable, simpler, more comprehensive and focussed information processing and control systems

Use of more reliable components and systems

Reduction of maintenance shutdown duration

Design for minimal system radioactive contamination and corrosion

Design for easy component replacement

Design for robotic maintenance

Design for easier monitoring of component state (e.g., brittleness)

Reduction of scheduled shutdown frequency

Increase fuel burnup and refueling interval

Design components for longer useful lives

Increase system design margins to allow more easily for accommodation to new adverse information enhancing plant availability. The latter roughly fall into the categories of making plants more reliable, more tolerant of failures and of new information regarding adverse phenomena, and easier to repair and maintain.

Operation and maintenance costs:
Table 6.1.5 summarizes several design measures which would reduce the cost of maintenance operations or avoid the real need for them altogether. Measures to reduce occupational doses associated with equipment replacement and other maintenance operations are of particular importance.

Areas of High Priority Interest

Without a comprehensive survey of utility options, an attempt to specify the highest priority design goals among those discussed above would be premature. However, among the most probable candidates are the following:

- reliability improvements;
- simplification;
- refinement of plant safety criteria;
- optimal plant sizing;
- minimization of maintenance and equipment replacement shutdown durations and costs and of radiation dose burdens.

As these areas are expected to be of special importance in any LWR innovation effort, they are discussed individually in this section.

Table 6.1.5

Factors Related to the Costs of Maintenance Operations

Maintenance avoidance

- Use of low corrosion materials and ultra clean water
- Use of simpler, more compact water filtration systems
- Better component monitoring to identify maintenance needs

Radiation dose reduction

- Use of modular, easily replaced components
- Design for remote automated maintenance
- Avoidance of use of easily activated, long-lived materials

Reliability Improvements. Improvements in plant reliability can be achieved in several ways, including the following:

- Making reduced demands upon control systems (human and mechanical);
- Improved components;
- Better-trained, more competent operators aided by more helpful information systems;
- More robust, simpler systems;
- Improved monitoring and maintenance of sensitive components.

The tools necessary to address the various means of reliability range from Probabilistic Risk Analysis (PRA) and historical records -- for identifying the most sensitive components for monitoring and attentive maintenance -- to common sense (which perhaps is not so common) in the case of more robust, simpler systems. The difficulty in attaining high reliability in current plants is that it typically does not appear as an explicitly identified, unambiguously quantified, design goal. Rather it is approached through setting design, manufacturing, and performance specifications for the various station components.

Quality Control (Q/C) during construction of the station is intended to ensure that these requirements are met. Associated with the Q/C program -- especially regarding components of the plant which are safety-related (which may in itself be a dangerous concept) -- is the Quality Assurance program (Q/A) which is intended through a combination of inspection and documentation, to be able to demonstrate upon inquiry that specified procedures have been followed. The Q/A system is punitive in that its immediate goal is to avoid the sanctions attendant upon failure to conform to the required procedures. It promotes the goals of reliability improvement to the degree that it aids the Q/C program in assuring that the various component specifications are met. In return, the Q/C program can promote high reliability only to the degree that the specifications which it enforces reflect that goal. Reliability is only one of several design goals which may be promoted in component requirements specification. Among others are component capacity maximization and the ability to function in hostile environments.

What is needed in this area is a far more comprehensive approach to the achievement and maintenance of a high level of reliability. This goal will be much easier to meet if it is made a more explicit component of the plant design and operational requirement than is currently the case. There are difficulties involved in incorporating such a goal into the set of goals pursued by the groups who typically design power plants -- the architect/engineer firms and the utility equipment divisions. These groups are different from those who usually operate the completed power stations. While the former may attempt to respect the spirit of a requirement for high plant reliability, unless this requirement can be stated in an unambiguous quantitative fashion it will remain difficult to ensure that it is being met adequately. The formulation of that statement is not obvious but is so important that it will require considerable care and attention.

The most important ways in which reliability improvements can be obtained using existing plants is to examine their operating histories with the goal of identifying and remedying the leading contributors to unreliability. (See Table 6.1.3). It is also important to ensure that the broadest and most current data are employed in comprehensive PRA's oriented toward identifying the leading sources of unreliability.

Plant Simplification. The idea of plant simplification is attractive but vague as there are limitless ways in which such a goal may be approached. There exist no clear criteria or methodologies to guide a designer to the most effectively simplified plan. The generalprinciples for plant simplification are the following:

- Re-examining the function of each plant system and component in order to identify those which are not essential and/or to determine whether this function could be satisfied adequately by a different, simpler system. As an example,it has been suggested that the PWR soluble boron reactivity control system should be eliminated, since this would significantly reduce the complexity and number of components in the plant piping system. The price of doing this would be a possible reduction in the uniformity of the power distribution in the core (though this could possibly be avoided through core redesign). Whether this simplification is worthwhile depends upon whether its net estimated benefits are positive, and, if so, whether they are greater than those of other simplifications which could be achieved with the same resources.

- Identification of unnecessary or avoidable interactions between plant systems and components. Avoidance of such interaction is essential for minimizing the potential of common-mode failures and of needless propagation of the consequences of improper operation of a particular system.
- Adoption with each system and component of a requirement that it be demonstrated that the function of that system be necessary or beneficial. If it is necessary its function should be provided in the simplest way possible. If its function is unnecessary but beneficial, then the net evaluated benefits of the function (including allowance for the costs of unreliable operation) should be substantial. This principle is expected to be especially important in assessing the best way of meeting plant safety requirements, since the designs of many current plants reflect a consistent pattern of using design "add-ons" in order to meet evolving safety criteria which became imposed after initial planning was completed.

Some examples of possible LWR design simplifications are listed in Table 6.1.6. The list is not intended to be comprehensive. Indeed, current designs offer a very large number of possibilities. Identification of the most important opportunities in this regard must rely substantially on the judgement and experience of the persons most familiar with the existing plants.

Table 6.1.6

Examples of Ways in Which Simplication Could be Used to Improve Plant Design

Design Change	Benefit
Elimination of soluble control systems in PWRs	Reduce piping system complexity possible failure modes, costs.
Use of oversized PWR pressurizer.	Increase range of secondary system events which could be tolerated without use of primary control system. Possible elimination of PORVs.
Use of large volume, inert atmosphere containments.	Elimination of need for hydrogen generation mitigation system Reduced need for atmospheric sprays, fan coolers.
Use of signal multiplexing, possibly with fiber-optical transmission.	Reduced use of signal cable, simplified signal network, greater network reliability, smaller number of containment penetrations.
Elimination of safety system bunkering.	Reduced dependence of safety system upon reliable functioning of auxiliary cooling system and service water systems.

Refinement of Safety Criteria. From the beginning of the nuclear power era, and especially during the past fifteen years, the safety requirements of LWRs have increased significantly from one year to the next. As a result, no plant in existence today reflects an optimized, comprehensive approach to the satisfaction of current safety requirements.

The typical evolution of a safety concern is from the status of a new issue, the validity of which is uncertain, through a period of investigation and of resistance by the nuclear industry to its inclusion in the set of mandatory safety concerns, to a final definition of the concern and of a set of allowable design solutions. In principle, a nuclear power plant license applicant remains free to challenge both the basis of the safety requirement and the identified solution, but the high costs and low probability of being successful is enough to deter such challenges in the great majority of cases. In practice, the identified solutions are incorporated into new plant designs and as backfits on existing plants as a path of least resistance in meeting the requirements of the regulatory authorities.* The result of this process has been the development of an extensive literature of Regulatory Guides and Regulations codifying the issues of concern and their accepted solutions. This vector of safety concerns is not formulated so as to reduce public risk most efficiently using available sources; it is not internally consistent; and it is not permanent (reflecting, among other factors, the persistent existence of a list of unresolved safety issues).

* The major notable exception to this rule in recent years is the post-Three Mile Island "source term" re-examination of the treatment of off-site radioactive releases in nuclear plant accidents.

Even for a single hypothetical accident the existing regulations are inconsistent. As an example, in a specified loss-of-coolant accident the safety system is required to maintain the fuel (and its contents) in a stable condition, the containment system is required to perform effectively assuming that core stability is not maintained, and the off-site emergency response plan is to be formulated as if the containment failed to function properly. This cascading of assumed failures in the determination of individual system design requirements is justified on the basis that it provides a residual defense against accidents more extreme than those specified as a deterministic design basis.

The result of this pattern of evolution is, in a given plant, a set of approximately integrated systems designed to address the set of safety requirements which had been codified at the time of the plant's initial design, and a set of systems added later to address newly-evolved safety criteria. The arbitrariness and incompleteness of the safety criteria and the high costs of the resulting systems suggest that a rationalization is desirable which could also provide for increased safety through more efficient use of available resources.

Important areas in which substantial improvements could be achieved by refining safety criteria are listed in Table 6.1.7. In all cases the purpose would be to increase the realism of the criteria and postulated accidents in order to maximize the degree of safety obtained with a given commitment of resources. This would involve identifying the greatest contributors to potential risk, and relying to the fullest justifiable extent on

Table 6.1.7

<u>Possible Opportunities for Rationalizing and Refining Plant Safety Requirements</u>

<u>Phenomenological modeling</u>

Emphasize the consistent upgrading of the realism and accuracy of models used for safety analysis in order to reduce needless and possibly misleading "conservatism."

Maintain long range industry and governmental research programs to provide the basis for modeling improvements. This would constitute a departure from the current practice of withdrawing resources from areas in which regulatory authorities have removed pressure, but where a substantial refinement of requirements remains feasible.

<u>Seismic and dynamic loads</u>

Reformulate design basis earthquake to be more realistic.

Design systems conservatively but dynamically and consistently, not statically.

Permit allowable system response to include tolerable plastic deformation.

Redefine postulated failure modes so as to be more realistic (e.g. focus more upon small pipe cracks than upon double-ended ruptures of major pipes).

Combine temporarily-varying component loads stochastically rather than statically -- in worst combination.

<u>Risk evaluation</u>

De-emphasize the current set of highly-stylized, deterministically specified design basis accidents.

Undertake programs to provide much greater data base to improve PRA accuracy and versatility and use PRA to identify the leading contributors risks, as a guide to redefine design basis accident formulation.

Couple PRA to NRC Safety Goal prescription and prescription of design basis accident set.

/continued

Table 6.1.7 (continued)

Emergency planning

Shift focus to realistic rather than conservative planning basis.

Shift the public response emphasis of such planning from evacuation to protective actions based upon seeking shelter and minimizing dose received.

Emphasize formulation of extensive quick-access libraries of analyses prior to accidents to facilitate rapid accurate diagnosis of accident conditions.

Safety system design

Shift from need for active quick-acting systems to passive systems required only to respond slowly.

Emphasize use of natural convection for emergency heat removal.

Introduce passive primary, containment safety heat sinks.

well-understood physical phenomena in designing systems to provide protection against those risks.

It is important to recognize that if a revision of plant safety criteria along the lines suggested is indeed to occur, the electric utilities will have to take the lead. Although the nuclear plant supply industry (vendors, architect-engineers, component suppliers, etc.) would presumably benefit from such an effort in the long run, its incentives to act are weaker.*

Optimal Size of Plant. A factor not tied directly to LWR innovation but which significantly determines the qualitative design options available to a plant is that of the plant's generation capacity. The range of sizes (900 to 1250 MWe) of plants currently available from the nuclear vendors is the result of past attempts to capture economies of scale. Whether such economies can easily be captured is debatable, with evidence being available pro and con. However, the reality is that the available size range is rather narrow, and possibly not well suited to the needs of the utility industry. In the past large plants could easily be accommodated into the expansion

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* This is reflected in the organization of the current IDCOR (Industry Degraded Core) effort, which is largely utility-financed with only weak nuclear-vendor architect/engineer leadership. Other evidence consistent with this view is provided by the deliberate decoupling of advanced LWR system design efforts currently being pursued in Japan by Westinghouse and General Electric from the domestic nuclear plant market. A strong motivation for this is to avoid subjecting the advanced designs to review by U.S. regulators, both to protect the designs from aspects of American requirements which are considered to be unjustified, and to avoid introducing additional safety innovations into the American regulatory system for fear that their existence will be used as a rationale for requiring that existing plants incorporate them in their designs. Thus, even though such innovations could permit a better use of resources in addressing safety requirements, the vendors do not at present perceive it to be in their interests to introduce them into the United States.

plans of the typical rapidly growing utility. However, current conditions of uncertain demand growth, of difficulties in keeping nuclear construction projects on schedule, and of high costs of capital increase the financial risk to a utility embarking upon such a project. These factors of uncertainty and possibly large cost may argue in favor of use of smaller plants, for which the attendant risks are smaller.

If this is the case, the range of feasible plant innovations becomes significantly greater than if another larger LWR is to be built as the next unit in the system. Among the possible areas of innovation are those of greater use of modular construction, prefabrication, standardization, and use of pre-licensed multi-unit sites than is possible with 1000 MWe plants. Such plants could also presumably be constructed more rapidly, have more complete initial designs and pose less individual risk (due to lower core fission product inventories) than larger plants. Whether such plants should be pursued depends upon the situation of an individual utility and upon the evaluated net benefits of these factors of small-scale economy. Because it is at least plausible that optimized smaller plants could be attractive, it is important that future planning consider LWR use in this mode.

<u>Minimization of Maintenance and Component Replacement Costs and Associated Radiation Doses</u>. As the current generation of LWRs ages, utilities are experiencing an increasing burden of maintenance and equipment replacement costs and attendant radiation doses. The most dramatic instances have occurred at several plants where unanticipated steam generator repairs and replacements have been necessary. These and other tasks have been much

more costly and difficult than they would have been had the need for them been anticipated at the design stage. Designing for such operational problems in future plants should thus be an important objective. The keys to minimizing these problems are the following:

- Designing components as easily exchanged modules.
- Designing for easy component access and transfer out of the containment building.
- Designing for rapid work by humans and for access and compatibility with robotic repair devices.
- Use of frequent maintenance and component monitoring to prevent and detect deterioration prior to failure.
- Use of low activation and low corrosion materials and environments to minimize production of radioactive contaminants and to prevent component failures.

The opportunities for improvement in this area seem substantial. That they have been neglected previously provides a further indication of the need for greater utility involvement in future LWR design specifications.

Improvements to Existing Plants

An effort to refine the designs of new LWRs would also be expected to provide benefits to existing plants. The main limitation upon potential design changes to existing plants is that imposed by the additional costs of backfitting. In many cases these additional costs can be so large as to outweigh the potential benefits of the change.

The main areas where the benefits of new plant changes should also be easily translated to existing plants include the following:

- Improved reliability through increased component monitoring and use of more comprehensive and active information systems.
- Operation for reduced corrosion and radiation field buildup.
- Fuel management to increase refueling intervals.
- Rationalization of safety requirements in order to permit relaxation of current safety system performance requirements.

Thus, an effort to refine the design goals of new LWRs can also be expected to provide substantial benefits for existing plants, although the scope of such benefits would be more limited than for the new plants. The corollary to this is that a plant redesign effort focussed upon improvements to operating plants would provide only limited benefits for the refinement of new plant designs.

Proposed LWR research

It is proposed to carry out a research project at M.I.T. in conjunction with a consortium of leading utilities with the ultimate goal of developing a design specification for an advanced LWR which would come closer than present systems to meeting future utility needs. The research group would consist of several M.I.T. faculty and staff (including the project leaders) and also several full time technical staff assigned by the participating utilities to the project. The duration of the project would be approximately 5 years, and the M.I.T. research effort would be funded at a level between $0.8 and $1.6 million annually. A fuller description of the goals, organization, personnel requirements, budget, and schedule of this project is presented in Section 8.1.1.

The M.I.T. researchers would contribute to the continuing work of the group in defining the design goals, and would also individually conduct research - typically working with graduate student research assistants - needed to advance the understanding of possible LWR design improvements.

This work would build upon the traditional nuclear engineering research efforts at MIT and upon particularly relevant current research projects such as those in the following areas:

- Thermal hydraulics for operational and safety analysis
- Nuclear power station operational simulation and advanced information systems
- Seismic design refinement
- Behavior of crucial component materials
- Robotics
- Design of small, modular LWR and HTGR power stations
- Reliability of key operational and safety systems
- Historical trends in power plant design

More detailed examples of continuing investigations at M.I.T. which would be relevant to this work are presented in the appendix at the end of this section, to provide an idea of the scale and diversity of these research efforts.

6.2 The PIUS Reactor Concept

The "Process Inherent Ultimate Safety" (PIUS) reactor is a radically redesigned LWR concept proposed by Kare Hannerz of ASEA-ATOM in Sweden.* It has been designed from the outset with intrinsic safety in mind, and has an inherent capability of self-protection against loss of fuel integrity during severe transient or accident events.

Figure 6.2.1 and Table 6.2.1 illustrate the key features of this reactor concept, which include:

- enclosure of the entire primary system in a large PCRV filled with pressurized water; the hot primary system is immersed in a pool of cold borated water to which it is hydraulically coupled at two fluid interfaces (6 and 7 in Fig. 6.2.1). The remainder of the primary system is separated from the cold pool by low pressure ducting, and special steel gauze insulation.
- use of an immersed main circulating pump to dynamically balance circuit pressure drops during normal operation such that the interfaces are stabilized; under upset conditions the cold borated pool water is drawn into the core, shutting it down.

The large coolant inventory is sufficient to guarantee approximately one week of "walk-away" safety before external intervention would be needed to insure the continuation of shutdown cooling.

* K. Hannerz, "Towards Intrinsically Safe Light Water Reactors," ORAU/IEA-83-2(M), February 1983 (and revision of April 1983).

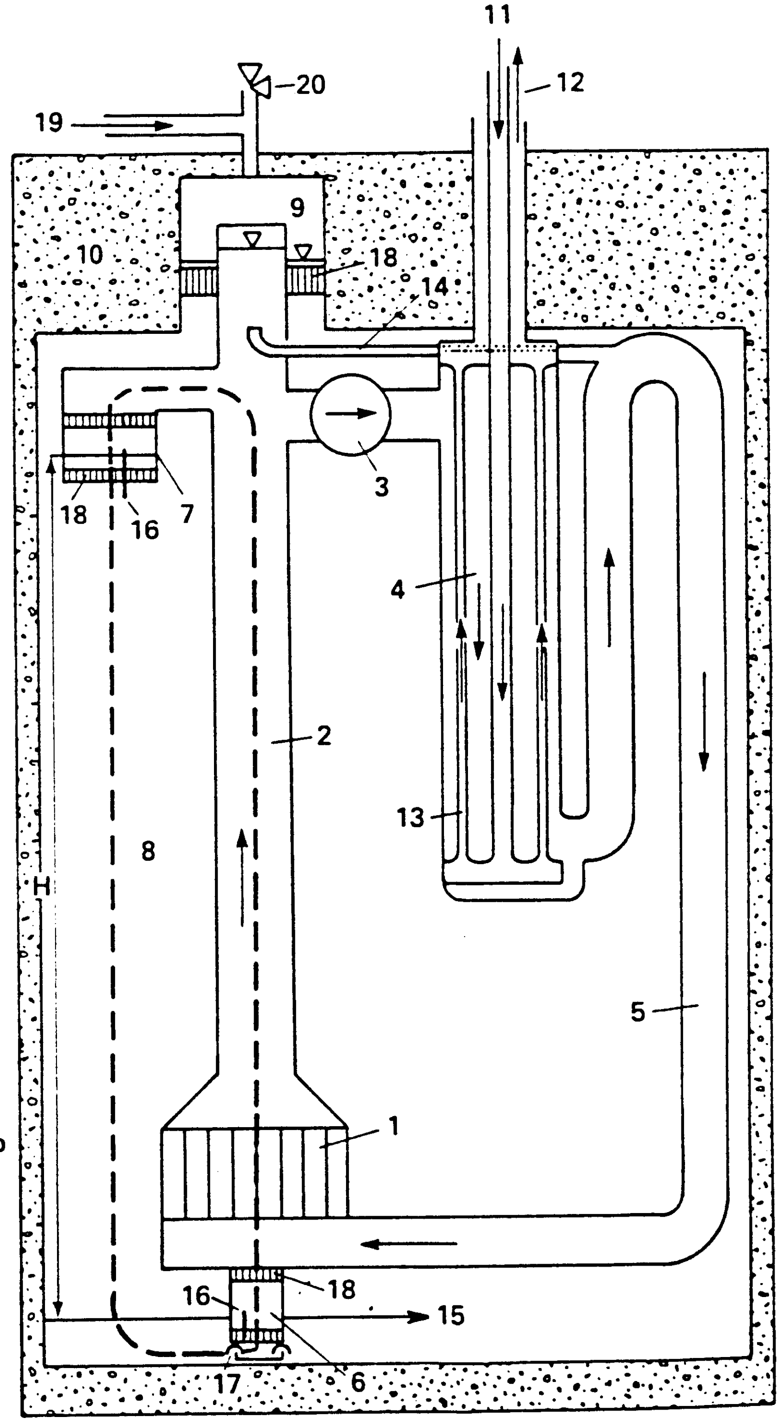


Fig. 6.2.1 Schematic of PIUS reactor conceptual design

Table 6.2.1

Representative PIUS Design Parameters

Power Rating:	
Thermal	1600 MW
Electrical	500 MW
Thermodynamic Efficiency	31.3%
Size of PCRV	43 ft i.d. 90 ft o.d. 100 ft I.Ht. 200 ft O.Ht.
Core Power Density	70.1 Kw/ℓ
Core Height; Diameter	6.46 ft; 12.60 ft
Operating Pressure (core exit)	1307 psia
core ΔP	6.44 psi
Core Inventory	68.4 MTHM
Design Burnup	30,000 MWD/MT
Steam Generators	once-through (boiling in tubes)
Fuel Assemblies	16 x 16 rod array 0.482 in o.d 0.600 in pitch, 2.82% enrichment, 193 assemblies
Number of:	
Steam Generators	4
Main Coolant Pumps	2

Several man-years of effort have been invested by its proponents in evaluation of the accident response of the PIUS system, and to date it has successfully withstood all postulated insults. The full roster of events has not yet been fully explored in depth, however, and questions still remain concerning such aspects as susceptibility to thermal shock, and instabilities during a depressurization transient. In addition to these low probability scenarios, more evaluation is needed in the area of design and performance of the once-through steam generator, and with respect to overall system maintainability. In this latter regard, in particular, it should be noted that the PIUS design is an evolving concept, and hence should be assessed in the form of its most recent embodiment.

Overall, the prospects are encouraging that PIUS may fulfill the goals for which it was designed -- greater assurance against severe core damage, hence against hazard to the public health and to utility financial integrity.

Nevertheless, there are some major uncertainties which caution against an unqualified endorsement of the PIUS concept at this state of its development.

(a) If a containment building is required (as we speculate would be the case in the U.S.), then a PIUS unit would be slightly more expensive than a conventional LWR. Indeed, the potential impact of the licensing process on all aspects of system design is a major concern: the reintroduction of complexity could further compromise economic prospects.

(b) Hydraulic stability is not yet fully proven -- the reactor may shut itself down too frequently without just cause; the time to recover from these scrams, particularly those involving rod drop, appears to be sufficiently long to raise concern over maintenance of a high capacity factor. Computer simulation studies and experimental work in the area of hydraulic response are currently in progress.

(c) Apart from its greater simplicity, PIUS does not appear to be significantly better suited to other innovations which may be necessary to promote a renewed interest in nuclear systems by either U.S. utilities (downsizing, modularity, accelerated construction) or long-range strategic planners (uranium utilization and low fuel cycle cost). However, 30 years' worth of in-vessel storage is available for spent fuel -- a safeguards advantage, and overall the reactor is less vulnerable to sabotage and earthquake damage.

The investment in time and money required to prove out this concept is not insubstantial, despite the fact that maximum use is made of available PWR/BWR technology. PIUS is sufficiently different that construction and operation of a prototype unit (which might be as small as 20-50 MWe) is called for, preceded by 3-5 years of preliminary engineering analysis and experimentation. Commercial plants could therefore not be committed to for a decade or more. R & D costs, including the prototype unit, over this time period would probably total on the order of one billion dollars. In addition to arranging for a commitment of funds at this level, it will probably be necessary to find a U.S. industrial sponsor/licensee willing to act as vendor for this system. Recent indications in the trade press are that at least one major utility (TVA) and an architect-engineering firm (Burns and Roe) have more than a passing interest in the PIUS concept.

Finally, PIUS may be viewed as one end of a spectrum of redesigned LWRs. For example, more forgiving versions of the BWR can also be devised. Some compromise design hybrid between PIUS and more conventional reactors may turn out to best satisfy the (as yet unquantified) consensus goals for next-generation units.

The recommended strategy for the MIT effort is to start from current LWR design configurations and progress, through an evolutionary process, toward a concept or concepts best suited to next-order units for the U.S. utility market. Although individual faculty members at MIT may contribute to the PIUS (and other) programs, it would appear that the PIUS evaluation process is both well founded and well underway, and hence that, even apart from philosophical differences, a concerted effort in this area under the aegis of the MIT Reactor Innovation Project would be redundant.

6.3 High Temperature Gas-Cooled Reactors

The properties of the high temperature gas cooled reactor (HTGR) follow in a particularly direct way from the microscopic properties of its fuel and coolant. HTGRs utilize small uranium oxide or carbide fuel kernels coated with pyrolitic carbon and/or silicon carbide dispersed in a graphite moderator cooled by helium. The use of a refractory moderator and helium coolant allows operation at high outlet temperatures; most designs eliminate metallic structures entirely to facilitate such operation. The large thermal capacity of the graphite core and the large negative temperature coefficient of reactivity make HTGRs relatively insensitive to reactivity insertion and to loss of coolant accidents. Low fuel concentration and direct embedding of the fuel particles in the high conductivity graphite matrix yields low peak fuel temperature and small thermal gradients. These generic properties plus others to be discussed later have sustained interest in HTGRs even though they employ a fuel cycle that was once considered to put them at a severe commercial disadvantage.

There are two major subspecies of HTGR depending on whether the reactor core is composed of randomly packed fuel/moderator spheres ("pebbles" of approximately 6 cm diameter) or of monolithic graphite (typically 80 cm high, 36 cm hexagonal "prismatic" blocks) with included fuel and coolant zones. In either case, the bulk graphite making up most of the core is mixed with resins and then molded at high temperatures and pressure to form extremely robust fuel elements. The U.S. program has, for the most part, considered reactors with monolithic cores. The German program pioneered and continues to exploit the pebble-bed concept.

There is a large body of experience with these reactors. HTGRs have been operated for over 15 years in the United States, the Federal Republic of Germany and the U.K.[1] The Japanese are currently building a small (50 MWth) experimental unit. At the moment, the German program is the most active.

In the U.S., experience with HTGRs has been based on operation of the 40 MWe Peach Bottom 1 plant from 1967 to 1974 and on the 330 MWe Fort St. Vrain plant that has been in commercial operation since 1976.[2] Although limited at first by many difficulties, the Fort St. Vrain plant is now demonstrating the high fuel integrity, low radiation exposure to plant personnel, and ease of reactor control that is typical of the HTGR design. The Fort St. Vrain operation has achieved 771°C primary loop temperatures and 538°C steam temperatures. A flow diagram on a plant schematic for Fort St. Vrain is shown in Fig. 6.3.1. Extensions of the Fort St. Vrain technology, including the use of prestressed concrete reactor vessels (PCRVs) has been the subject of designs for both large electricity-producing HTGR plants and for cogeneration of steam and electricity for process applications. Figure 6.3.2 shows a typical large (1000 MWe) design utilizing a prismatic element monolithic core in a multicavity PCRV. This particular design would produce steam of 1000°F and 2415 psi, suitable either for process heat or for cogeneration.

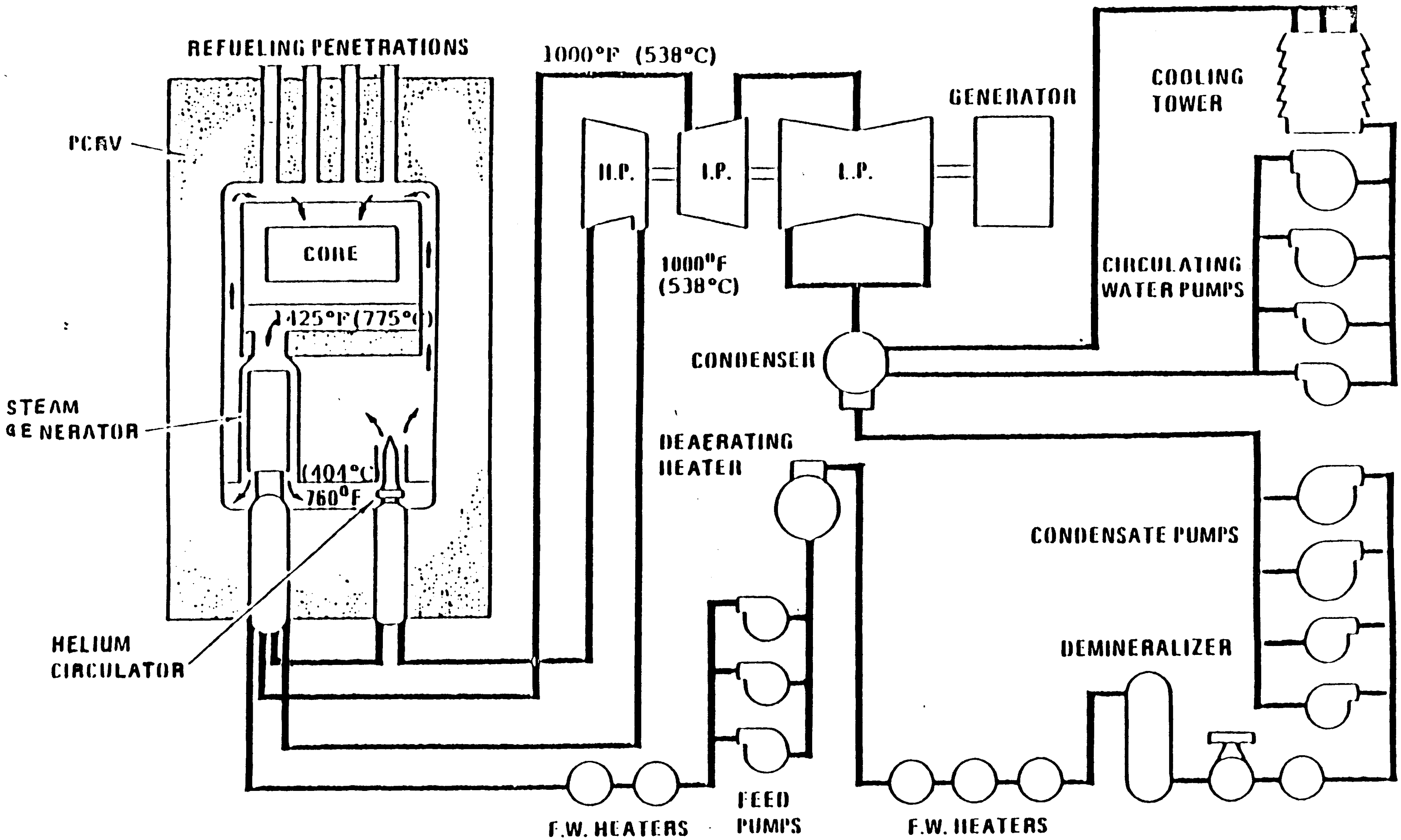


Figure 6.3-1. Fort St. Vrain Flow Diagram

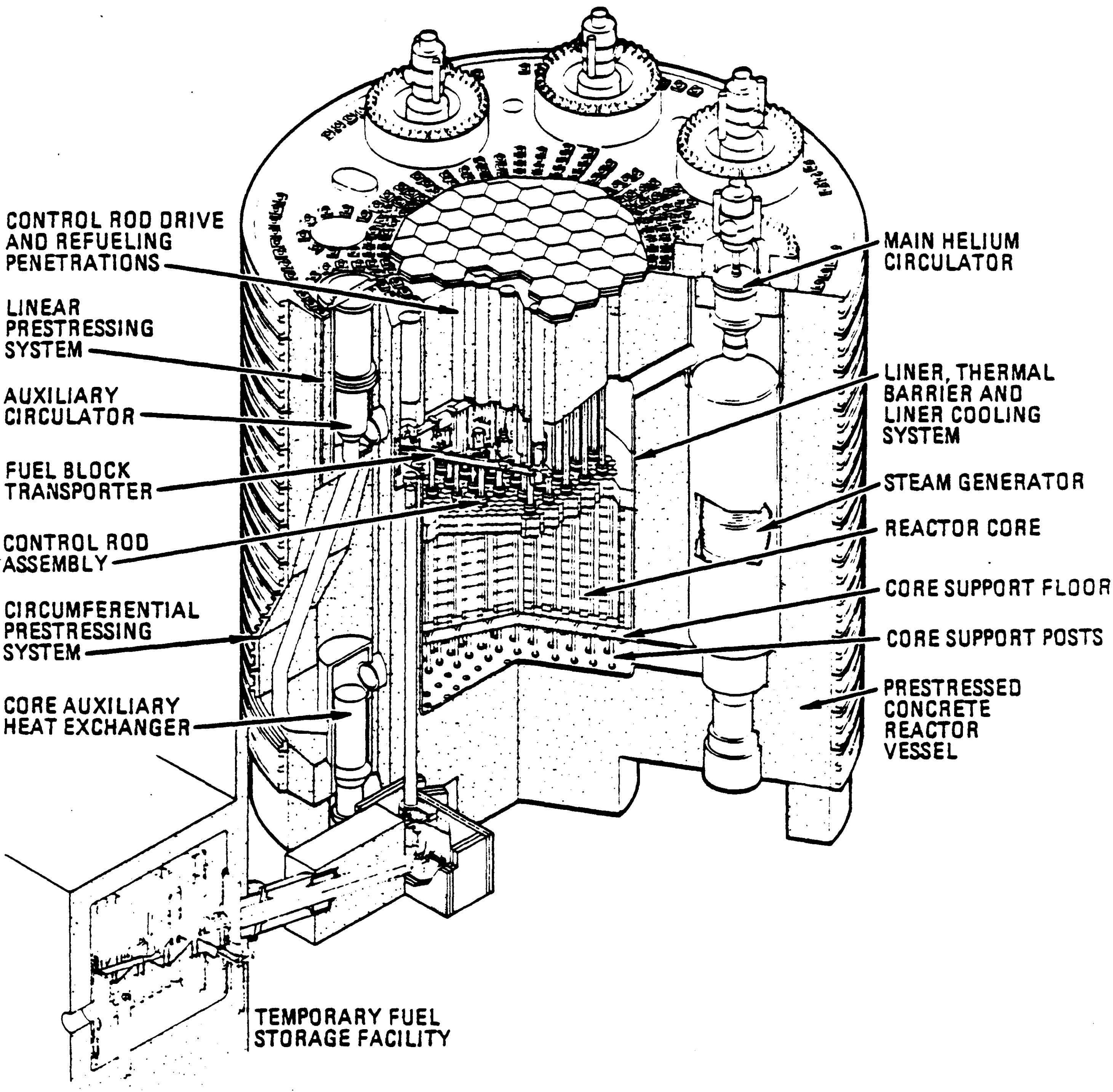


Fig. 6.3.2 2240-MW(t) HTGR-SC/C

The German HTGR experience is based primarily on the AVR; a 50 MWth research reactor that has been in nearly continuous operation since 1968.[3] AVR, designed to produce very high temperatures, operates at a coolant outlet temperature of 950°C. A much larger (300 MWe) pebble-bed demonstration reactor (THTR)[4] is effectively completed and is scheduled for initial criticality in October 1983. THTR is designed for 750°C outlet helium temperature. There is also a continuing series of studies of 350-500 MWe units supported by a consortium of German utilities because units of this size are particularly well suited to the German power grid. It is possible to design small HTGRs for negligible fission product release (i.e., peak core temperature below 1600°C) even in the event of a depressurized loss-of-coolant flow accident. Recent work at KFA and KWU has concentrated on designing units of largest possible size that still retain the negligible fission product release attribute of smaller units. Figure 6.3.3 is a cross-section of a 350 MWe annular pebble-bed core design under study at KFA.

The Germans consider pebble-bed fuel technology to be essentially fully developed -- over two million pebbles have been utilized to date -- and are concentrating on various reactor implementations of the technology. Of course, HTGRs promise high efficiency electrical generation but the high temperature capability is also of particular interest for gasification of Germany's large supply of hard coal and lignite.

The fuel cycle for the HTGR was initially conceived with the idea of using highly enriched uranium fuel combined with thorium. This cycle offers a high conversion to U-233 and a beneficial reactivity lifetime. However,

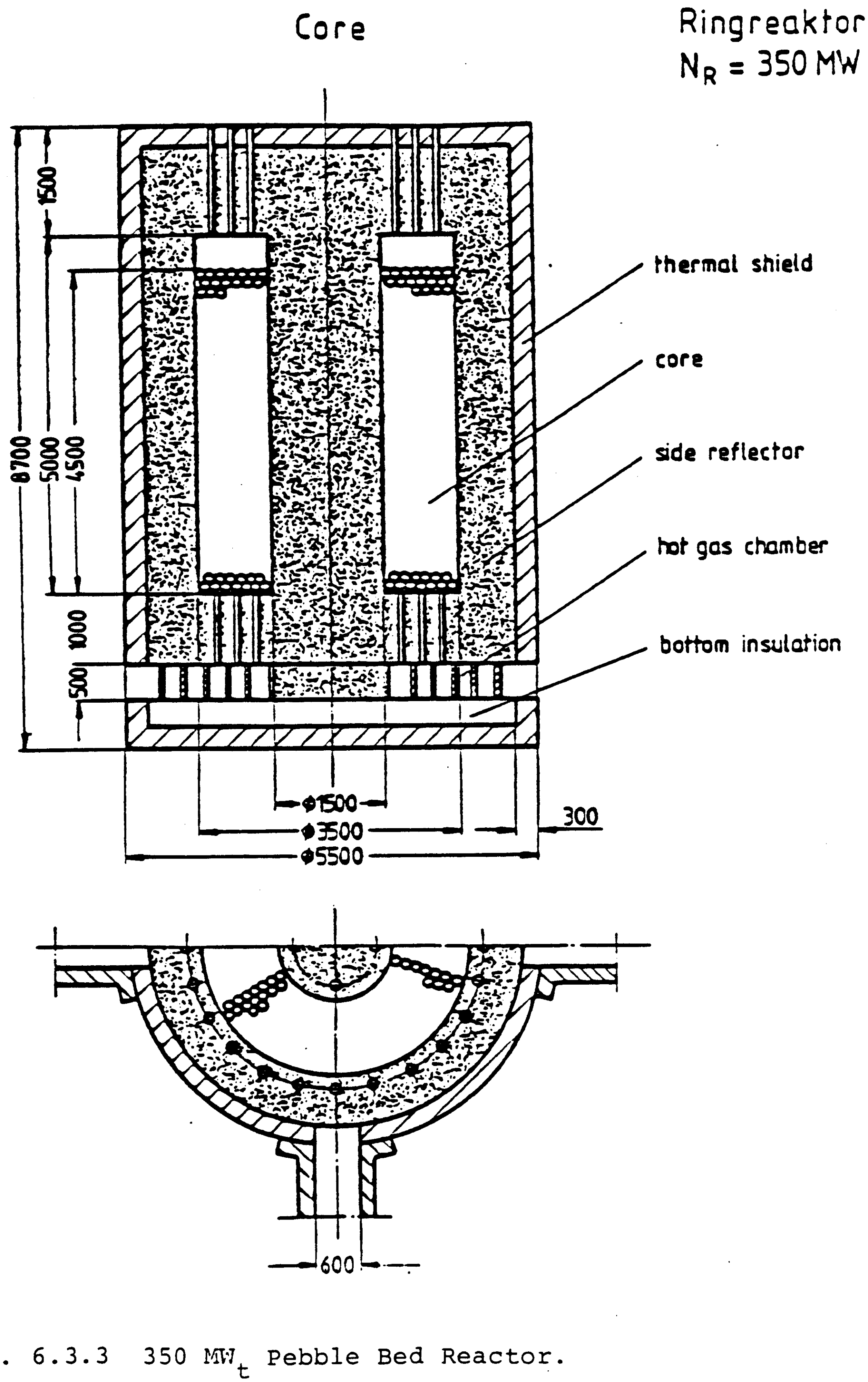


Fig. 6.3.3 350 MW_t Pebble Bed Reactor.

recent emphasis has centered on a low enriched uranium once-through cycle similar to that employed in present day LWRs. The use of low enrichment fuel has been thoroughly demonstrated in the Dragon[5] reactor in England as well as in the AVR.

Thus, commercial HTGRs exist and advanced designs have been developed that have the potential for producing process heat as well as high efficiency (40%) electricity production. From a technical point of view, in comparison with present day LWRs, the large HTGR has advantages in increased safety margins, lower radiation exposures to operating personnel and higher thermal efficiency. However, these advantages have not been strong enough to overcome the initial increased financial costs and uncertainties that arise in making the first large commercial HTGR.

As a general conclusion, one can observe that although on paper the HTGR looks very acceptable, it has not been possible to prove that large HTGR plants have fewer financial uncertainties than present-day LWRs and this, combined with the uncertainties in such areas as U.S. licensing, has delayed any commercial use of the HTGR for electricity or process heat.

6.4 Advanced High Temperature Gas-Cooled Reactors

The large thermal capacity and high limiting temperatures of the HTGR core offers the possibility of significant, qualitative advantages with respect to the LWR. Because these advantages are most evident in smaller units, there is considerable interest in modular systems of 250-300 MWth size and a substantial body of literature is available. Although very small special purpose (i.e., military) reactors have been studied for some time, the commercial potential of reactors of less than 100 MWth has not been adequately explored. This end of the size spectrum has, in principle, unique and possibly very valuable attributes. We will consider below the categories of "small" and "very small" HTGRs.

Recent HTGR studies in the U.S. and the FRG envisage a small number (4-8) of identical modules as moderately redundant power plant heat sources, or highly redundant (i.e., with capacity substantially greater than nominal output requirement) process heat sources. In these studies, individual reactor modules are designed to be as large as possible consistent with the constraint that none of the fission products be released in a depressurized loss of coolant flow accident. This design choice is made in order to take advantage of certain economies of scale and to reduce the complexity associated with the connection and simultaneous operation of individual reactors. The typical unit size consistent with this design approach is of the order of 250-350 MWth, although modular systems have been designed with 600 MWth units.

There is also another regime of operation for which the HTGR may be uniquely suited. In this regime of "very small" reactors the economy of scale of large individual units is replaced with the economies of serial production, shop fabrication, and standardization of very much smaller units. These units might be typically 50-100 MWth. In Section 6.4.1, we discuss the first class, called here small modular HTGRs. In Section 6.4.2 we consider the somewhat more speculative very small HTGR. In these sections we emphasize the potential benefits that we believe exist and point out several of the unresolved problem areas. In Section 6.4.3 we present a list of research topics which concentrate on one or another of the problem areas or which quantify the potential benefits. We note that each of the individual topics appears capable of resolution within a five year time frame. We believe that pebble bed HTGRs are a particularly fertile area for intensive study. Such systems have certain unique properties and share in others not available in combination elsewhere: high inherent safety, high efficiency, extensive data base, flexible fuel cycle, promise of high reliability plants, etc. The reduced financial risk of small modules will be obvious to the utilities but perhaps equally important, the publicly perceived risk might be substantially smaller because the relatively low cost of a single modular unit will facilitate proof testing in a variety of circumstances of units identical to those intended for commercial operation. Experimental verification is inherently more convincing, and certainly more accurate, than mathematical simulation of complex systems. This fact will not be lost on either the public or the utilities.

Other institutional barriers are similarly avoided. For example, the relatively low cost and existing technology base might allow new vendors to enter the arena. The combination of small size and simplicity would allow small utilities to contemplate nuclear systems, facilitate load growth matching and greatly simplify the problem of financing nuclear systems.

6.4.1 Small Modular HTGR Systems

Introduction

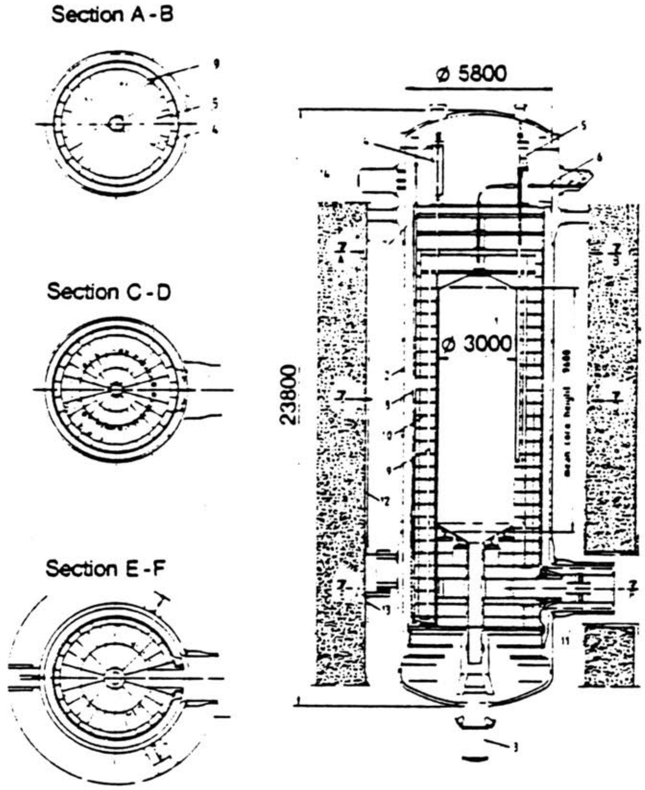
The concept of a nuclear power plant composed of several small modular walk-away-safe units is becoming increasingly attractive. During this time of slow growth in the demand for electricity and difficult new plant financing due to escalated costs of large plants, it is reasonable to consider smaller incremental investments in modularized components. Thus, a plant might be initially small but designed for expansion in steps commensurate with availability of financing and increases in electric demand. It should be noted that each individual module represents smaller financial risk from unforeseen difficulties or accidents and that redundant modular systems can be designed for unit maintenance or replacement without loss of plant availability.

In particular, it appears valuable to consider the pebble-bed high temperature gas-cooled reactor (PB-HTGR) as the nuclear reactor module. In Europe, this type of reactor is called simply the high-temperature reactor (HTR) but for consistency in this report will also be referred to as the

HTGR. This reactor type utilizes a well-developed fuel, pyrolytically coated fuel particles inside small (approximately 6 cm O.D.) graphite balls, cooled by helium gas. These fuel "pebbles" can be circulated or reshuffled through the core, thus providing on-line refueling capability. The combination of high temperature and on-line refueling results in a high efficiency system with potential for high plant availability. These characteristics, together with certain inherent safety capabilities, have been demonstrated by 15 years of operation of the AVR plant in Germany.

Description of a Small HTGR (approximately 200 MWth)

The principle characteristics of small HTGRs include the integrated arrangement of all primary loop components in a steel reactor pressure vessel. The modular HTGR described by Lohnert and Reutler is a good example of the class (see Fig. 6.4.1a). As shown, the flow of coolant gas goes through the core to the steam generators located to the sides of the core at an elevated level, with the control rods and back-up shutdown mechanisms located only in the graphite reflector. The back-up shutdown system is composed of smaller absorber spheres which can fall into the channels in the reflector if needed to assure shutdown. The fuel elements have shown good performance up to peak operating temperature of 1250°C and maximum burn-up of 160,000 MWD/t.[6] At present, a "once-through cycle" is contemplated where the spent fuel is loaded into cast-iron vessels (50,000 pebbles in each) for storage and disposal cycles. The core is designed so that temperatures do not rise above about 1600°C even in a hypothetical accident such as loss of all active coolant systems. As long as the fuel temperature remains below



1 Pebble-bed
2 Pressure vessel
3 Fuel discharge
4 Small-pebble-absorber system
5 Reflector rod
6 Fuel loading
7 Thermal shield
8 Core vessel
9 Coolant channel
10 Side reflector
11 Hot gas duct
12 Wall-cooler
13 Insulation
14 Concrete cell

Fig. 6.4.1a HTR-Module (Ref. 7).

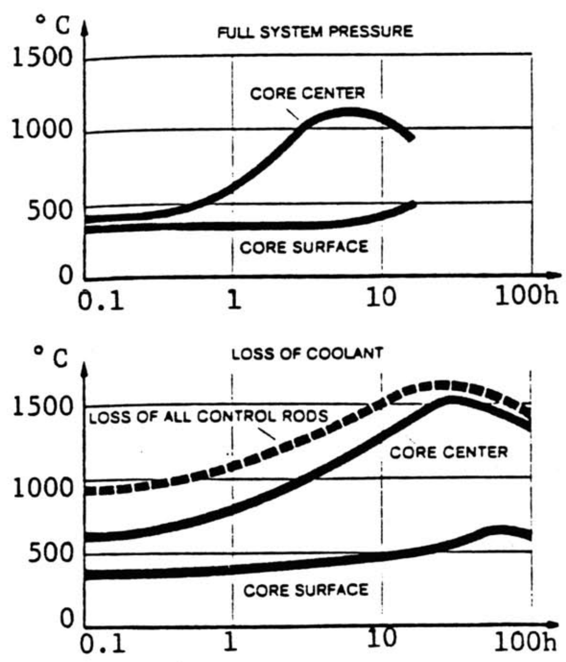


Fig. 6.4.1b Time Dependent Maximal Fuel Element Temperatures Under Accident Conditions (Ref. 7).

1600°C, there is no appreciable release of fission products from the pyrolytically coated fuel particles.

For after-heat removal following a reactor shutdown, three possibilities are provided:

(1) Normally, after-heat removal is performed through the operating water-steam cooling loops, by natural convection of the He-coolant;

(2) In case the water-steam system is not available, the after-heat can be removed on the secondary side through a 4-loop auxiliary cooling system.

(3) If all systems fail the after-heat can be removed primarily by radiation and conduction from the pressure vessel to a 2-loop concrete cooling system which is located in the inner concrete layer of the surrounding containment structure. In case of loss of concrete cooling, the reactor will still be safe (although the concrete might be damaged). Also, if the control rods do not shut the reactor down, the low excess reactivity with on-line refueling and the high negative temperature coefficient can cause the reactor to be safely shut down during the heat-up. The fuel temperature limit of 1600°C will not be exceeded (see Fig. 6.4.1b).

Some Additional Features of Small HTGRs

The following features of small HTGRs should also be noted:

- Emphasis can be made on simplicity for high degree of reliability and ease of operability to minimize support staff requirements. In addition, the reactor can be more easily automated than other concepts because of the single phase coolant, large core heat capacity and simple fuel form.
- The dose level to operating and maintenance personnel is very small compared to other nuclear reactor systems (low coolant and component activity).
- The reactor offers operational flexibility, fast high-power ramping and rapid start-up from part load conditions.
- High efficiency of the HTGR minimizes thermal heat rejection.

Finally, although the primary objective in this project is to study systems for electric generation, it should also be noted that small HTGRs represent an excellent alternative to big plants in the heat market.

Modular Utilization of Small HTGRs

In the case of reactor components like steam generators, circulators, etc., it is usual to connect several smaller units rather than one large unit to obtain a required large power. But the possibility of increasing the power output by connecting several "reactor cores" has not often been seriously considered, i.e., a power source was never thought of being generated by several individual "cores." It is now being proposed that a plant with high total energy capacity could beneficially be composed of more than one reactor in the same containment building, a so-called "modular system."

In big nuclear power plants, the cost of engineered safeguards is becoming higher and higher. To avoid this complication, simplification of the safety design has been pursued for small HTGRs. Moreover, it may also be possible to reduce the specific cost of the modular concept by increasing the number of small reactors to be built in the same reactor containment building. Increases of fuel-cycle and capital costs would contribute only moderately to the overall plant costs. Even for the medium-sized power-plants the fuel-cycle costs and capital costs for the reactor-core contribute only approximately 1/7th to the overall costs for electricity production.[7] Hence, it seems possible to separate, or modularize, the heat source in analogy to the common practice of separating the heat sink without too-heavy acost penalty and to design a fully modularized HTGR system. This process leads to a reduction of the financial risk of introducing high-temperature reactors into the market since plants of various sizes and applications can be derived from one common basic concept.

Some Features of Modular HTGRs

- The specific plant costs of a modular HTGR could possibly be lower than the specific plant costs of a large prismatic HTGR of conventional design at least up to the medium power range. Major reasons are:

(1) Use of standardized, prefabricated steel-pressure vessels.

(2) Simplifications in the core design due to the limited core diameter (e.g., reflector control rods only).

(3) Reduced or no need of redundant decay heat removal systems or of emergency power supplies inside the primary system.

(4) Elimination of need for gas-tight containment building (this possibility is still debatable).

(5) Standardization and serial (possibly shop) production of all essential components. A further cost reduction would be attributed to a reduction in plant construction time.

(6) Shutdown systems can be designed on simple principles since the requirements are determined by requirements for cold shutdown only.

(7) No active auxiliary cooling systems are necessary inside the primary system.

(8) Licensing procedure once established should be accelerated over the LWR since the safety-related analyses are easy to verify and

because many of the design basis accidents can be demonstrated experimentally (e.g., the AVR supercritical loss of coolant flow tests).[8]

6.4.2 Very Small Modular HTGRs (50-100 MWth)

At the smallest end of the modular reactor size spectrum, it is possible to conceive of standardized reactors that are built in serial production by centralized shops, and shipped over the road to serve as interchangeable thermal modules for electrical and process heat applications. Such a scenario, so much at variance with the existing one, is attractive only if the economy of serial production is comparable to economy of scale in large units and if the individual units are, and are perceived to be, incapable of releasing radioactive fission products.

Reactors designed for this regime of operation must have certain mechanical and operational features. The major mechanical constraint is obviously size, with maximum core diameter limited to 4-5 meters to allow shipment of complete units. Other desirable mechanical features include the use of a steel containment vessel, simplified internal structure, and the use of external (reflector) control systems. The minimal operational requirements are simplicity of control, stability of operation, and most importantly demonstrable safety under all postulated accident conditions, i.e., an individual module must remain undamaged in the event of a fully depressurized accident with full reactivity insertion. Because there will be many modules in a single system, simple fuel handling techniques are highly desirable, although on-line refueling will be unnecessary (single units can be taken off line with only a small impact on availability). It is this combination of features that would allow the simultaneous operation of many identical units without the rapidly mounting costs associated with intereacting control circuits and multiply-redundant safety systems.

The plant concept suggested above essentially describes the parallel operation of a large number of highly simplified "user-friendly" reactors. The technology is not new; in fact, there is a substantial data base. Prototypes of such reactors exist in the form of TRIGA (exploiting the properties of Uranium-Zirconium Hydride fuel) and AVR. Of the two, AVR appears to have substantially more commercial potential. The required fuel fabrication technology is available both in this country and abroad, and, of course, the operating temperature is substantially higher.

With only modest redesign and extrapolation of demonstrated AVR technology, it appears possible to build a very small gas cooled reactor (VSGCR) that has the attributes of:

- demonstrable safety
- stable operating point
- simple fuel handling
- shop fabricability
- external (i.e., reflector) control
- low operating expense

VSGCRs designed for this mode of operation would have several disadvantages with respect to large HTGRs and even with respect to existing LWRs. First, the neutron economy is relatively poor in small systems because of neutron leakage. Second, coated particle fuel designs do not have a demonstrated reprocessing system. The first problem appears sensitive to design mitigation and should be a major focus of research in this area. The second disadvantage is probably inconsequential in view of the general trend to once-through fuel cycles and the time frame of this study.

The postulated multiple VSGCR plant design is not technology driven but is specifically designed to meet the minimum requirements for utilities considering purchase of a nuclear power plant: predictable cost and reliable operation. These attributes follow directly from the large scale shop fabrication of identical units and from the high order redundancy of a single power plant.

6.4.3 Suggestions for Future Study

The following topics address the major unresolved issues:

(1) The interconnection of several individual plants must be studied to develop optimum control strategies for smooth operation under expected normal conditions and anticipated modular system transients.

(2) The core physics of very small reactors must be studied with the goal of understanding and optimizing fuel utilization, controllability, and transient behavior.

(3) Further safety analysis studies should be made concerning water or air ingress, and work on impurity limits in the helium coolant as well as long-term effects of impurities needs to be continued. The development of combustion-resistant fuel pellets is of particular interest.

(4) Further research must be done on designs that can allow safe installation of a new module while other modules are running. This might include the use of robotics for installation, removal and remote maintenance.

(5) Further study must be done on designs to minimize component radioactivity and make well-spaced component installations for safety and ease of expected maintenance and eventual decommissioning.

(6) The auxiliary systems, such as cooling water loops, fuel loading systems, and gas purification systems, must be laid out largely independently for each individual plant in order to achieve an adequate reliability of operation and also to avoid larger consequences due to interactions in the case of accidents in an individual unit.

(7) The often-discussed question of movements within the pebble-bed core under vibrational excitation should also be studied. Major HTGR components should be investigated with respect to operation under seismic conditions.

(8) The containment system requirements must be studied and established.

(9) Work should be continued that is in progress in the U.S. (at GE and GCRA) and Germany on design and cost estimates of small modular HTGR systems.

The following are a selected group of topics proposed for a program at MIT.

- Comparison studies of prismatic versus pebble-bed modular HTGR designs including both safety and economic considerations.
- Core physics in very small reactors.
- Study of modular size optimization over the range of possible modular sizes.
- Study of computer control of the modular system including possible applications of robotics to maintenance.
- Study of the containment system requirements and designs for the modular arrangement.
- Assessment and study of improved modular steam generator designs.

The thrust of these proposed MIT projects is in the area of modular reactor systems studies. Although the potential benefits in this area seem substantial, the research to date on modular systems has been limited. Much work must be done in conjunction with groups such as the Gas Cooled Reactor Associates (GCRA) if utility interest is to strengthen. Implementation of the proposed work is discussed further in Chapter 8.

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3. Atomwirtschaft, May 1968 -- Entire issue devoted to AVR.

4. N. Kirch, W. Jansing and G. Lange, "Status and Aims of the Research and Development Programme for High Temperature Reactors in FRG," Proceedings, Gas Cooled Reactors Today, BNES London, 1982.

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6. R. Fisher and I. Weisbrodt, "HTR Development of Nuclear Processs Heat in the PNP Project," Proceedings, Gas Cooled Reactors Today, BNES London, 1982.

7. G.H. Lohnert and H. Reutler, "The Modular HTR, A New Design of High Temperature Pebble-Bed Reactor," Proceedings, Gas Cooled Reactors Today, BNES London 1982.

8. Ref. 1, p. 2-6 et seq.

6.5 OTHER CONCEPTS

During the early days of the nuclear power era, scores of reactor concepts were examined at the paper study level, and more than a dozen were pursued as far as the operating prototype stage. While this winnowing-out process may not always have been carried out on the basis of strict technical/economic merit -- particularly as we now view matters, having the benefit of several decades of experience -- this is largely a moot issue, since present realities require that near to intermediate term innovations be supported by a substantial framework of both base technology and plant operating experience. Thus, beyond the concepts already explored elsewhere in this report, only heavy water reactors and sodium cooled fast breeder reactors can muster the requisite credentials for serious consideration. This is born out by the roster of power plants, arranged by nuclear genotype, presented in Table 6.5.1.

6.5.1 Heavy Water Reactors

While a number of reactor concepts have been devised which employ heavy water as a moderator, the CANDU type reactor is by far the most highly developed system for central station power generation. Moreover, the impressive operating record of more than ten large units in Canada argues strongly for their consideration in any review of future nuclear options. In assessing their potential for the U.S. market we are also the fortunate

Table 6.5.1

Nuclear Power Plants (Operable, Under Construction, or on Order ($\geq$ 30 MWe), as of 6/30/80

TYPE (COOL/MOD.)		U.S.	WORLD
PWR	LWR(H_2O)	113 (63.5%)	280 (53.2%)
BWR		61 (34.3%)	119 (22.6%)
LWBR			
PHWR (CANDU)	(D_2O)		36
LWCHWR			2 (8.0)
HWBLWR			2
GCHWR			2
GCR	Graphite		36
AGR			15
LGR		1	23 (14.4)
HTGR		1	1
THTR			1
LMFBR (Na)		1	8
TOTAL UNITS		178	526
TOTAL GWE		171	400
TOTAL OPERABLE		74	229
GWE OPERABLE		54	125

KEY:
- PWR = Pressurized Water Reactors
- BWR = Boiling Water Reactor
- PHWR = Pressurized Heavy Water Moderated and Cooled Reactor
- LWBR = Light Water Breeder Reactor
- LWCHWR = Light Water Cooled, Heavy Water Moderated Reactor
- HWBLWR = Heavy Water Moderated Boiling Light Water Cooled Reactor
- GCHWR = Gas Cooled Heavy Water Moderated Reactor
- GCR = Gas Cooled Reactor
- AGR = Advanced Gas-Cooled Reactor
- LGR = Light Water Cooled, Graphite Moderated Reactor
- HTGR = High Temperature Gas Cooled Reactor
- THTR = Thorium High Temperature Reactor
- LMFBR = Liquid Metal Cooled Fast Breeder Reactor

Source: Nuclear News, Vol, 23, No. 10, August 1980.

beneficiaries of an in-depth study carried out for the DOE by Combustion Engineering Inc. as part of the NASAP effort.*

Table 6.5.2 summarizes some of the key features of the CE version of a pressure tube reactor and Fig. 6.5.1 shows a typical core layout for one of these units. Based on the results of their study and our own review, the following points appear particularly germane to present concerns:

(1) The CANDU is more capital intensive than the LWR, but capable of greater long term fuel economy; its higher inherent (and actual) capacity factor offsets part of the capital penalty.

(2) This concept appears to be less susceptible to whole-core involvement in hypothetical severe accident scenarios, but potentially more vulnerable to small LOCA events associated with features provided for, or the conduct of, on-line refueling.

(3) The CANDU appears to be no better suited than the LWR to implementation of the innovations under consideration in many quarters for revitalization of the nuclear option -- e.g., modularization, downsizing, simplification, augmentation of inherent safety features, and the like.

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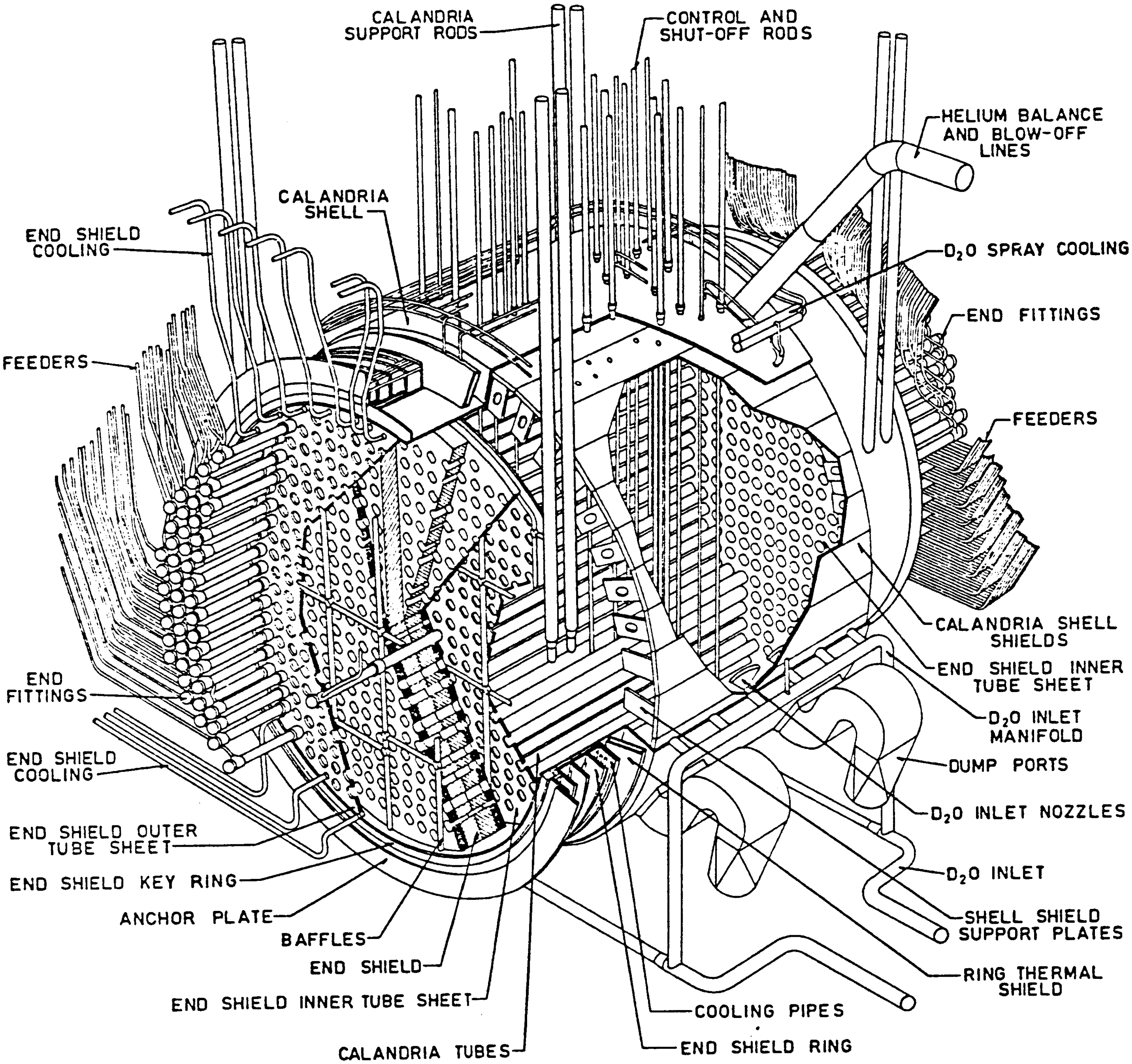
* N.L. Shapiro and J.F. Jesick, "Conceptual Design of a Large Heavy Water Reactor for U.S. Siting," CEND-379, Vols. I-IV, Sept. 1979.

Table 6.5.2

Proposed Features of U.S. PHWR (CEND-379)

Thermal Power Rating	4029 MW
Electrical Power Rating	1260 MW
Number of Primary Loops/Steam Generators	2/4
Number of Pressure Tubes	740
Moderator Inventory	414 Metric Tons D_2O
Fuel Inventory	166 MT
Fuel Enrichment	1.2 w/o U-235
Lifetime (30 yr) U_3O_8 Requirement	3545 tons
Lifetime SWU Requirement	927 MT
Discharge Burnup	19,750 MWD/MT
Capital Cost Relative to LWR	
Excluding D_2O	1.08
Including D_2O	1.34
Busbar Power Cost Relative to LWR	
Near Term (40\$/lb U_3O_8)	1.02
Long Term (100\$/lb U_3O_8)	0.94
RD&D Costs:	
Generic Technology	\$14 x 10^6
Demo/Lead Plant/Commercial Plant Safety & Licensing plus first of a kind engineering costs	\$100 x 10^6



ISOMETRIC VIEW REACTOR PICKERING

Fig. 6.5.1 Section through a representative CANDU reactor core

(4) Some consider that this type of reactor can offer more of a safeguards risk than the LWR, but this aspect is unlikely to be of much significance in the present context.

(5) Importation of this technology would involve complications which are greater than involved with those other systems for which a base of U.S. vendor/AE/utility (EPRI) support and experience already exists.

(6) As indicated in Table 6.5.1, several variations on the PHWR theme have also received some attention, differing from the main line effort chiefly in the choice of coolant (light water, boiling light water, and gas-CO_2); in addition, a small organic cooled prototype has been operated quite successfully in Canada. None, however, appear to be enough of an improvement to dislodge the basic CANDU concept from its preferred status among this class of reactors.

All-in-all, it is our conclusion that the CANDU system does not appear to be "sufficiently different" to justify serious hopes that it, or any evolutionary extension of its essential features, could radically alter the prospects of nuclear power in the U.S.

Many of the same caveats which have been cited with regard to the CANDU system can be applied to other D_2O moderated systems -- in particular, the pressure-vessel type HWR (e.g., ATUCHA*), and the spectral-shift PWR.** Again, they are not a sufficient departure from current technology to merit their substitution for the LWR -- it is preferable to pursue evolutionary improvements in the latter.

6.5.2 Fast Reactors

With the notable exception of Canada, all countries which have embarked on substantial nuclear power programs have targeted breeder reactor development as the focus of their long-range planning. In the U.S., the breeder program in general, and CRBR in particular, have born the brunt of much of the protracted debate over the role of nuclear power -- so much so that a detailed review of either breeder policy or technology would be superfluous here. However, approached with a fresh perspective, the LMFBR has much in its favor as an inherently preferable concept: low pressure operation and a superbly effective coolant being the most obvious. Nevertheless, apart from all the sound and fury of partisan debate, it is clear that the high capital cost of present designs and reliance on fuel reprocessing will considerably delay deployment of the breeder reactor in the present economic environment. Thus, one is led to inquire whether a version

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* ATUCHA II, Nuclear Engineering International, Vol. 27, No. 332, Sept. 1982.

** R.A. Matzie and G.P. Menzel, "Conceptual Design of a Large Spectral Shift Controlled Reactor," Vols. I & II, CEND-377, Aug. 1979.

of the breeder reactor more in tune with present needs can be devised; as the following comments suggest, some rather radical possibilities exist.

One obvious (albeit ambitious) target for plant simplification and cost reduction in the LMFBR is the elimination of the intermediate sodium loop. It is conceivable that this goal could be accomplished by re-introduction of the concept of the duplex-tube steam generator. While EBR-II employed duplex-tube units, this approach has not been followed in subsequent breeder designs. It is perhaps not entirely coincidental that EBR-II has not had a steam generator leak in over 17 years while leaks have been a common problem in LMFBRs built since. Indeed, some contend that the steam generator is the "achilles heel" of the LMFBR, and recent PWR experience does not strengthen the case for leak-free technology. It is encouraging to note that EPRI has recently supported a re-assessment of the duplex-tube concept,* (although still for use with an intermediate sodium loop).

The U.S. is in a particularly good position to move in this direction because its design emphasis of late has been on the "pipe" rather than the "pool" concept. Hence it should be easier to "replace" the intermediate heat exchanger by a steam generator; whereas it would not be safe practice to immerse a steam generator in the pool of primary sodium, as would be the case if the IHX of a pool concept design were substituted for.

A second major change in philosophy which could be considered is the option of starting up the first generation of breeder reactors using enriched

* EPRI NP-2316, "Component Development for Duplex Tube Steam Generator for Large LMFBR Plant," April 1982.

uranium instead of plutonium fueling.* (This in fact has been the practice in the majority of fast reactors operated to date, but while it is proven practice, it has always been regarded as an unwelcome compromise.) This would free the breeder from reliance on the concurrent development of a reprocessing capability and make good use of the near term surplus in separative work capacity and the cheap ore now available on the spot market.

Although fast spectrum cores in the breeding mode deserve primary attention, it is well to recall that sodium-cooled thermal reactors, using graphite** or zirconium hydride*** as the moderator, have been constructed (sodium has a microscopic thermal absorption cross section 40% smaller than that of water). Hence this basic reactor type is an extremely adaptable concept. Even if attention is restricted to fast systems, a varied menu of options is available; for example, one can design cores which do not require refueling for the life of the plant (e.g., 30 years).**** Thus we are led to

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* C. Braun and S. Golan, "A Strategy for Fueling Early Large Proptotype Fast Reactors," Trans. Am. Nucl. Soc., Vol. 44, June 1983.

** C. Starr and R.W. Dickinson, "Sodium Graphite Reactors," Addison-Wesley, Reading, MA, 1958.

*** R. Harde and K.W. Stohr, "A Sodium Cooled Power Reactor Experiment Employing Zirconium Hydride Moderator," 3rd I.C.P.U.A.E., Vol. 6, Geneva, 1965.

**** R.A. Doncals, G.J. Calamai, J.A. Lake, "Additional Considerations in LMFBR Core Design Philosophy," Trans Am. Nucl. Soc., Vol. 38, June 1981.

stress the need for a clear definition of design objectives; it is all too easy to compile specifications having a given generic reactor type in mind, but most difficult to decide what will sell -- both to utility executives and the concerned public.

Given the above considerations, it would be beneficial if sodium cooled plant designs were prepared emphasizing features (down-sizing, inherent safety, rapid constructability, lower cost) currently being touted for other reactor concepts, so that a directly comparable assessment can be made of the ability of the LMFBR to fulfill the same role proposed for other "advanced" or "inherently safe" systems. This exercise would also have the salutory effect of forcing the design community to fully come to grips with the problem of defining precisely what technical specifications must be met by these next-generation systems. Some of this re-thinking is already in progress; the initial focus has been, for the most part, on cost reduction.*

The major deterrent to re-programming the LMFBR is the overall cost and uncertain schedule. Certainly another zig-zag in the already tortuous path of U.S. breeder development will only lead to an increase in R & D costs and a further slippage in schedule. On the other hand, given present

**W.H. Arnold, et al., "Design Approaches to Achieve Competitive LMFBR Capital Costs," Power Eng., Sept. 1982.

circumstances, earlier deployment of a significant number of sodium cooled systems could be expected, since they would now be decoupled from a purely resource-driven scenario. Thus a re-targeted sodium system might be judged eligible for consideration within the time frame of current interest -- one to two decades. It is admittedly not as imminent as some of the other options we have considered, such as the HTGR, but still worthy of consideration, because it can naturally evolve into the breeding version for which it was originally conceived. If the breeder continues to be funded as a long-term option, the marginal cost of developing a near-term version may be rather small.

6.5.3 Other Reactor Types

Many additional reactor design concepts have been given serious consideration as recently as the late 1970s, during the NASAP and INFCE studies.* While the focus of these efforts was somewhat different than in the current MIT review, the extensive documentation developed in these programs provides a useful data base from which to launch our effort.

A critical re-reading of the NASAP and INFCE final reports has failed to identify other concepts (beyond those already discussed earlier in this and

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* (1) INFCE, Report of the International Nuclear Fuel Cycle Evaluation, IAEA, Vienna (1980).
  (2) "Nuclear Proliferation and Civilian Nuclear Power," Report of the Nonproliferation Alternative Systems Assessment Program (NASAP), DOE/NE-0001, Vols. 1-9, June 1980.

preceding chapters) worthy of consideration, with the possible exception of the molten salt homogeneous reactor -- because of its low pressure operation and low in-core fission product inventory. However, the practical aspects of reviving this reactor concept suggest that it not be given co-equal status with the other initiatives identified in this report. A major factor which could alter the status of the MSR in the long term (beyond that of primary emphasis here) would be the finding that molten salt systems are preferred technology for fusion reactor blankets, as some now contend.

In any event, the following observations are germane:

- Although the U.S. program to develop molten salt based systems has lapsed, there is a continuing (albeit modest) effort in Japan.* Hence, it might be possible to pursue this line of research on the basis of an international cooperative program.
- Although large (e.g., 1000 MWe) units are feasible, the MSR should be one of the more compatible concepts as regards the use of small modular units.
- Anticipated RD&D plus commercialization costs of the MSR, over a projected period spanning four decades, total between 6 to 9.5

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* (1) R. Ishiguro, K. Sugiyama and H. Sakashita, "Basic Studies for Molten-Salt Reactor Engineering in Japan," Proceedings of the Japan-U.S. Seminar on Thorium Fueled Reactors, Nara, Japan, Oct. 1982.
  (2) K. Furukawa, "The Combined System of Accelerator Molten-Salt Breeder and Molten-Salt Converter Reactors," (ibid).

billion dollars (1979$).** Hence, this concept is probably beyond the time frame and funding horizon of our current interest.

6.5.4 MIT's Role

The MIT Nuclear Engineering Department has a long history of participation in the development and analysis of advanced technologies, and it is anticipated that individual faculty members or teams of researchers will continue to fulfill this role. It is clear, however, that most of this activity should take place outside the project scope defined in this report because of the constraints imposed by the time frame of interest and limitations on the totality of available department resources. An effort should be made, however, to establish a more than casual level of liaison with both other major ongoing programs (CANDU, LMFBR) and important new initiatives (PIUS) to insure that different perspectives are fully appreciated and accounted for in developing a balanced evaluation of those alternatives which <u>will</u> constitute the primary foci of the MIT innovation initiatives. In turn, we may be able to encourage like-minded workers in these organizations to develop a parallel set of responses to the concerns motivating the MIT efforts.

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* ORNL/TM-7194, "An Assessment of Advanced Technology Options for NASAP," Sept. 1980.

Appendix to Section 6

1. Primary System Dynamic Response Modification

It has been suggested that PWR's could be made more tolerant of component and operational faults through increasing the characteristic time scale of system transient response to various initiating events. This would have the effect of allowing more time for the control system (both human and automatic) to respond correctly to unexpected events, and of increasing the envelope of events for which no control system action (e.g. activation of PWR pressurizer sprays to compensate for coolant expansion due to a temperature increase) would be required.

Among other options this could be accomplished through increasing the primary water inventory in the steam generators and/or increasing the pressurizer volume.

The implications of both of these design variations have been examined in order to quantify the time scales of variation of different primary system parameters of interest. This has been done by comparing similar primary system transients analyzed by the different PWR vendors in attempt to infer relevant design variation sensitivities. This has also been done by using a simplified PWR system model to analyze both the loss-of-feedwater-flow accident and the loss of load accident with scram.

Comparison of Design of Different PWR Vendors for the Steam Line Break Accident:

The designs of the Sequoyah plant by Westinghouse and the San Onofre plant by Combustion Engineering are summarized in Table 6.A.1. It is seen that the two plants are very similar in terms of electrical capacity but quite different in their designs. We have compared the primary system design differences in an attempt to determine whether reasons for the differences in the system responses to the steam line break accident (in which a steam generator outlet pipe breaks suddenly) can be inferred from the design differences. We have concluded that they can, and that the system could be made more tolerant of these accident by changing its design.

It is seen in comparing the Westinghouse (W) to the Combustion Engineering (CE) designs that the ratios of the times required for primary system depressurization to 1000 psia and for emptying of the pressurizer are all roughly equal to 2.6. These quantities are all governed by the rate at which the primary system liquid volume decreases due to cooling of the water in the damaged steam generator. The rate of liquid volume decrease is given by:

$$\frac{\partial V_L}{\partial t} = V_{SG}\,\beta\,\frac{UA\ \Delta T}{\rho C_p V_{SG}} = \frac{\beta}{C_p \rho}\,UA\ \Delta T$$

where

V_L = primary system liquid volume,

Table 6.A.1

Comparison of Westinghouse * and Combustion Engineering ** PWR System Designs and Steam Line Break Accident Responses

Item	Westinghouse	Combustion Engineering	Ratio: W/CE
Design Parameters			
Number of Steam Generators	4	2	2.0
Primary Liquid Volume per Steam Generator (ft^3)	1077	1880	0.57
Primary Coolant Mass (lb_m)	12,200	12,200	1.0
Heat Transfer Surface per Steam generator (ft^2)	51,500	90,200	0.57
Primary mass flowrate Steam Generator (10^6 lb_m|s)	33.5	74.0	0.45
Pressurizer Volume (ft^3)	1800	1500	1.20
Accident Response Times (s)			
Primary coolant pressure 1000 psia	28	10	2.8
Pressurizer empties	21	8	2.6
Reactor recriticality	25	20	1.25

* Abstracted from San Onofre FSAR
**Abstracted from Sequoyah FSAR

V_{SG} = primary system steam generator liquid volume,

β = water volumetric coefficient of thermal expansion,

UA = steam generator overall heat transfer coefficient,

ρ = water density,

C_p = water specific heat,

ΔT = temperature change from primary to secondary side of steam generator, and

t = time.

The primary liquid volume change needed to affect any of the quantities indicated is approximately proportional to that of the pressurizer. Thus, the ratio of the time scales for depressurization, or for pressurizer emptying, τ, is expected to be aproximately

$$\frac{\tau_W}{\tau_{CE}} \simeq \left(\frac{V_{Press-W}}{V_{Press-CE}}\right)\left(\frac{UA_{CE}}{UA_W}\right)$$

$$\frac{\tau_W}{\tau_{CE}} = (1.2)(2.06\,*) = 2.47.$$

The value for this ratio obtained from the respective plant FSAR's is 2.6, which is in reasonably good agreement with our crude estimate.

Similarly, the rate of insertion of reactivity into the reactor via flow of cold water from the damaged steam generator to the shut-down core will occur on a characteristic time scale, τ, given as

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* Inferred from FSAR results.

$$\tau \simeq \frac{V_{SG}\rho}{\dot{m}_{SG}},$$

where $\dot{m}_{SG}$ is the primary coolant mass flowrate through the steam generator.

The ratio of time scales for the two systems for bringing the shutdown reactor back to critical could then be expected to vary approximately as

$$\frac{\tau_W}{\tau_{CE}} = \left(\frac{V_{SG-W}}{V_{SG-CE}}\right)\left(\frac{\dot{m}_{SG-CE}}{\dot{m}_{SG-W}}\right)$$

$$= (.57)(2.2) = 1.26$$

The value obtained from the FSAR's for this ratio is 1.25.

From the foregoing we may conclude that the dependence of the system response to component design variations may be assessed by relatively crude analytical methods - for the steam line break accidents - and that the system response times can be increased substantially by increasing the volumes of the steam generator and/or pressurizer.

Analysis of the Loss of Feedwater Flow Anticipated Transient Without Scram (ATWS):

A computational model has been constructed at M.I.T. by Nuclear Engineering Department faculty and graduate students for simulation of a set of PWR system transients. The model and its use in an analysis of the Loss-of-Feedwater-Flow ATWS is summarized in a recent paper, enclosed in Annex A. This model has been used to examine the sensitivity of the system response to this accident to pressurizer and steam generator design volume changes of the system response to this accident.

The base case used in these analyses is that described in Annex A. Separate simulations were performed in which the volumes of the steam generator and of the pressurizer were varied. For most system dependent variables the sensitivity to these design changes is found to be effectively nil. The greatest sensitivity observed is that of the coolant pressure, for which the results are shown in Figs. 6.A.1 and 6.A.2. Even for this parameter the transient response sensitivity is weak.

These results can be explained by examining the progress of the accident. In this accident secondary coolant flow (feedwater) ceases and the reactor fails to scram but rather continues to generate power. As a consequence the primary coolant ceases to be cooled in the steam generator and continues to be heated when it flows through the core. As the coolant temperature in the core increases the correspondingly impaired neutron moderation results in a reduction of the core reactivity, and consequently of the core power.

The effect of increasing the primary side volume of the steam generators is to decrease the rate of temperature increase of the primary coolant and correspondingly to decrease the magnitude of the subsequent coolant

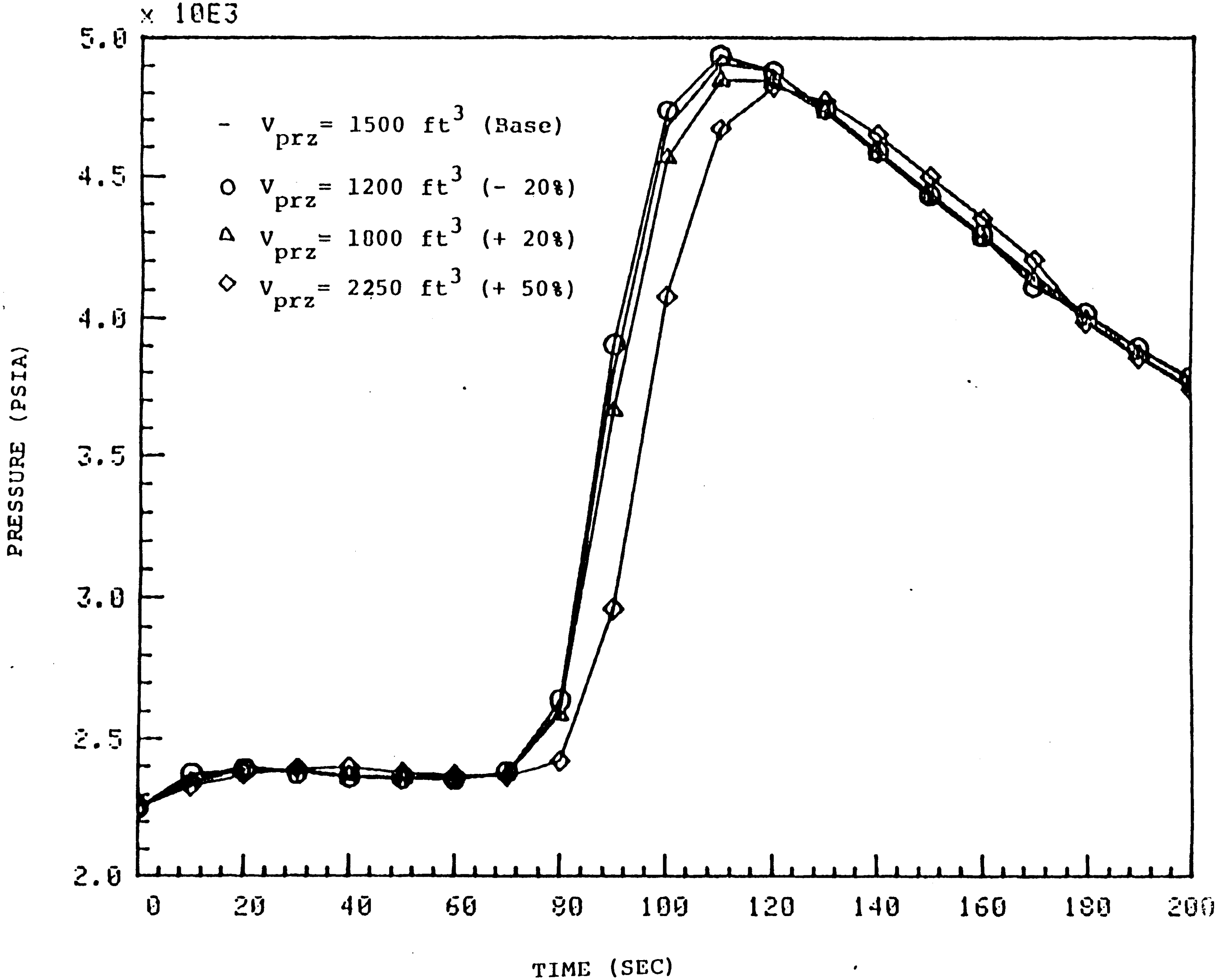


Fig. 6.A.1 Sensitivity of Primary System Pressure to a Change in the Pressurizer Volume for a Loss of Feedwater ATWS

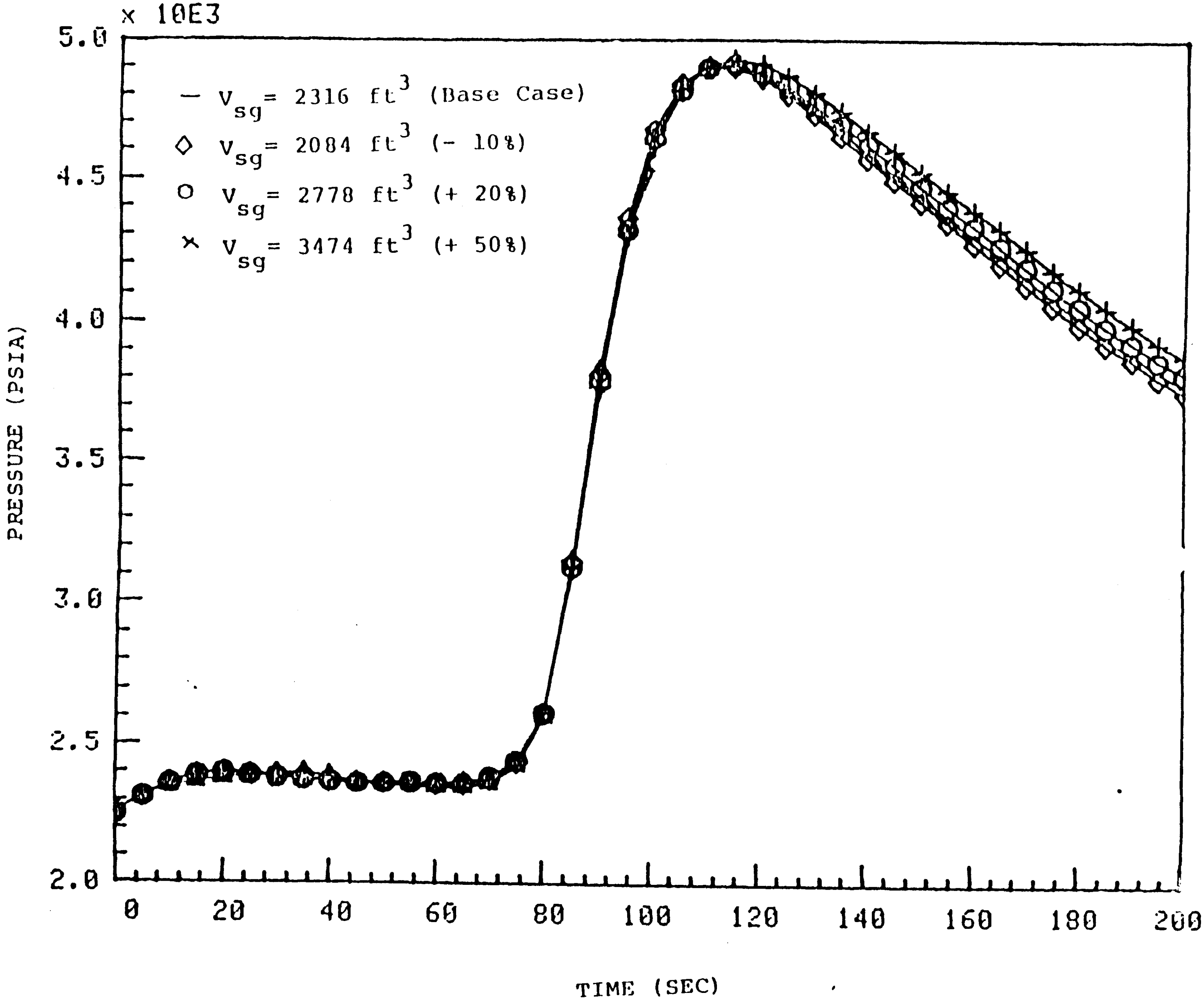


Fig. 6.A.2 Sensitivity of Primary System Pressure to a Change in the Steam Generator Volume for a Loss of Feedwater ATWS

temperature-related reactivity decrease in the core and of the subsequent power reduction. The net result of these coupled interactions is one of counterbalancing effects so that the primary coolant temperature, fuel temperature and core power histories are approximately the same for steam generators of various sizes.

The effects of pressurizer size variations are similarly small. During this accident, the pressurizer control system functions too slowly to have a conspicuous effect. The existence of the pressurizer becomes manifest mainly when the pressurizer becomes filled with liquid (due to coolant thermal expansion), resulting in a rapid subsequent coolant pressure increase. This behavior is seen in Fig. 6.A.1, however its pressurizer size sensitivity is small.

In this case the major design innovation which can provide a significant change in the system response is a change in the core moderator coefficient of reactivity. It can be seen from Annex A that the peak coolant pressure increases rapidly as the moderator coefficient decreases. Thus, this accident is different from the steam line break accident in that the system response is relatively insensitive to the component volumes. However, the responses to both accidents are sensitive to the core moderator coefficient values.

Loss of Load With Reactor Scram Transient:

The loss-of-load transient has also been analyzed using the model described previously. In this accident the turbine load is lost, secondary steam is directed instead to an atmospheric steam dump uponactuation of a relief valve and the reactor is scrammed. The secondary system is unable to act as an adequate heat sink for the primary coolant, resulting in primary coolant heating and expansion. In this process the pressurizer pilot operated relief valves (PORV) open at a pressure of 2400 psia, and the pressurizer sprays are activated at a slightly lower pressure.

The sensitivity of the system pressure response to pressurizer size variation is summarized in Fig. 6.A.3. It is seen for this case that the rates of both pressurization and subsequent depressurization can be decreased by increasing the pressurizer volume. By increasing the pressurizer volume from 750 ft^3 to 3000 ft^3 the rate of depressurization can be decreased from 0.13 psi/s to 0.06 psi/s.

In this instance as in the steam line break accident the primary system response time can be significantly increased by increasing the system thermal inertia.

Summary

The important lessons from this work are that it is feasible to perform university-scale research which can provide insight into the value of available reactor design innovations, and that it is necessary to investigate the full implications of a possible innovation (i.e. in different cases) in order to assess its value correctly. As is seen with the options of changing the steam generator and/or pressurizer volume significant increases in the

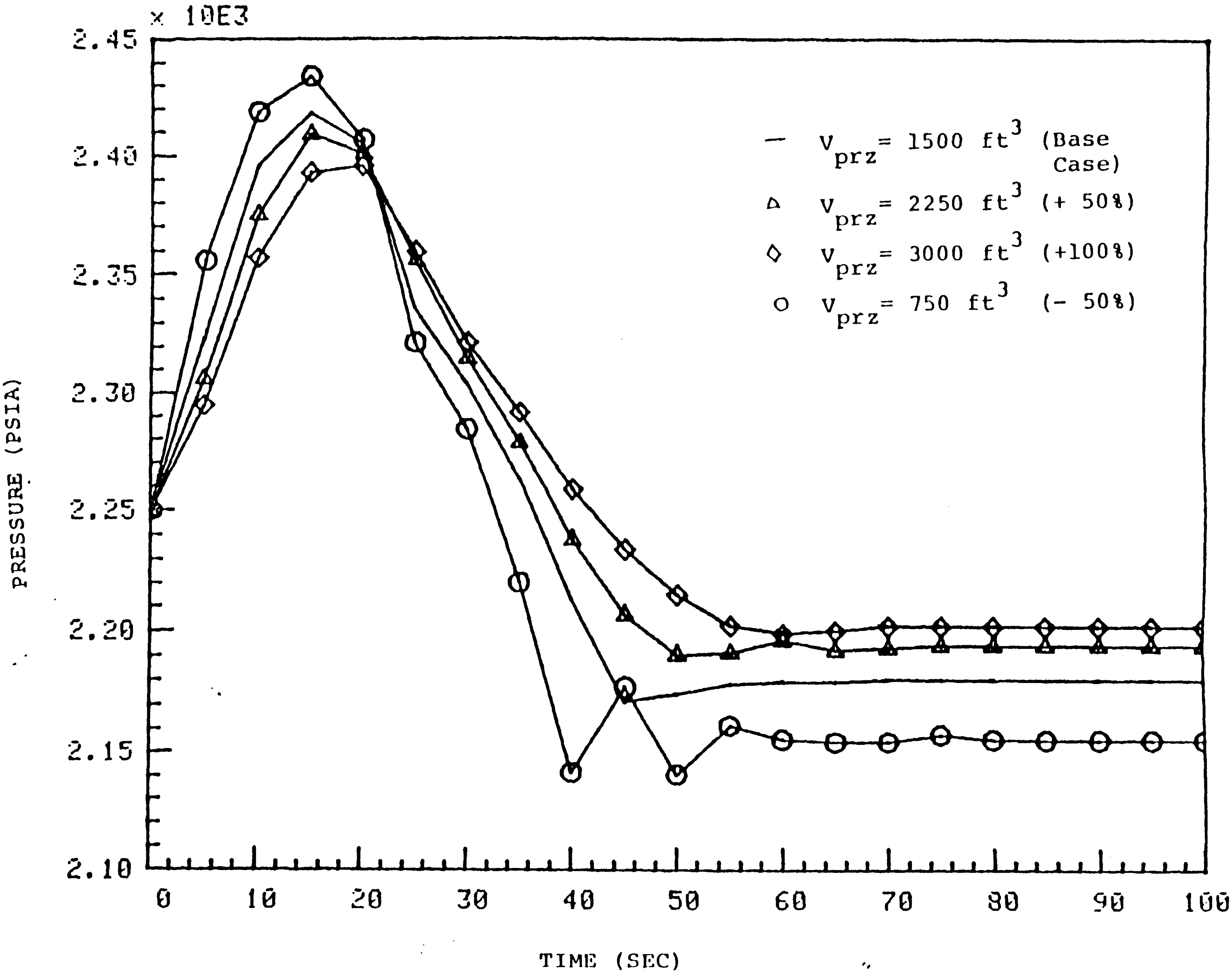


Fig. 6.A.3 Sensitivity of Primary System Pressure to a Change in the Pressurizer Volume for a Loss of Load Transient with Scram

system response times may be obtained for the steam line break and loss of load accidents, but the same design variations will provide benefits which are negligible (at best) for the Loss-of-Feedwater-Flow ATWS.

2. Investigation of Stress-Corrosion Cracking in Nuclear Steam Supply System Materials

The behavior of nuclear steam supply system (NSSS) materials in corrosive stressful environments has belatedly become recognized as being of primary importance in meeting goals of component longevity and power plant relability. Past failures of steam generators, core structural components and fears of future failures of reactor coolant piping have all involved stress corrosion cracking as the primary failure mechanism.

A research program at M.I.T. investigating the causes and cures of materials susceptibility to cracking in such environments is described in a recent paper appended as Annex B.

In this work the effects upon selected NSSS nickel-based alloys of hot working and annealing in permitting them to survive long-duration fatigue loads in various test environments are being investigated experimentally. Work to date has indicated that beginning-of-life heat treatment is important in extending the fatigue resistance of the materials tested.
A significant dependence upon both the amplitude and frequency of the fatigue loadings has also been observed.

3. Advanced Instrumentation and Control

Among the possible LWR improvements which could be pursued in the proposed project are those involving development and implementation of advanced instrumentation and control systems (see Table 6.1.7). Work currently underway at M.I.T. is directed toward this goal, and has included a demonstration of the feasibility of limited automatic reactor control. This is an initial step in the development of a set of "real time" subsystem models which may be used for analytic verification of reactor signals, for system state analyses and ultimately for reactor control. It is also an example of the class of university research which could provide substantial benefits in thisarea of LWR innovation. This work illustrates development of an innovation which can be initiated via technology-oriented research at a university prior to commercialization. Improvements in instrumentation and control have potential cost benefits in terms of reducing operator errors, increasing plant availability through early detection of pending problems and prevention of transients which could challenge the reactor safety system and which could cause thermal cycling of plant components.

An internally funded project concerned with Advanced Control has been in progress in the Nuclear Engineering Department for some time. This joint project between M.I.T. and CSDL (Charles Stark Draper Laboratory) is based on the following goals and concepts:

- Technology now exists, based on aerospace developments and applications, that can greatly enhance the instrumentation and control systems for nuclear plants (and also fossil plants).

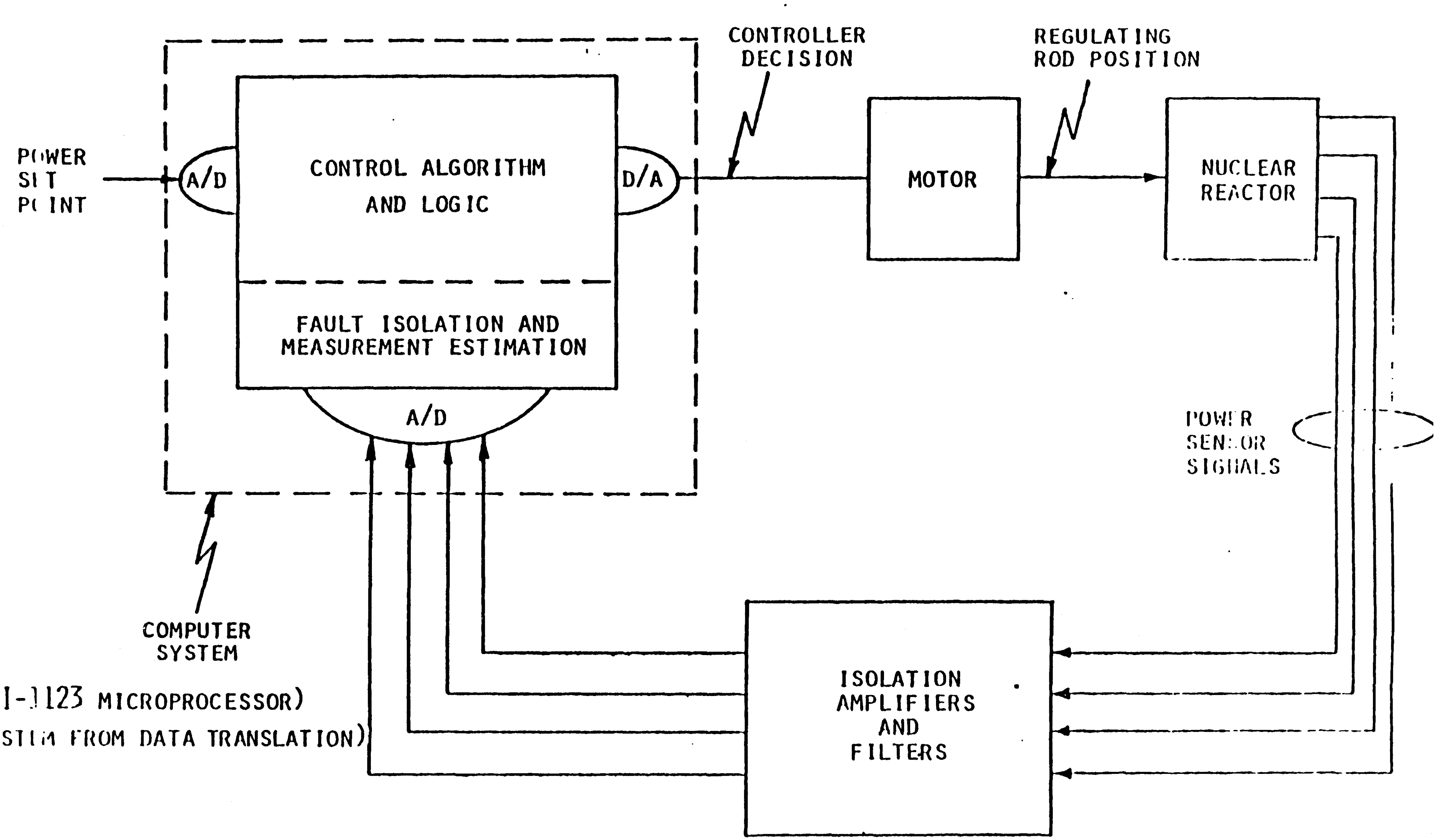
- The technology involves utilization of digital computer systems; specifically, the use of distributed mini- or micro-processors with fault-tolerant central processors connected through multiplexing systems for distribution and collection of information. The output includes validated control signals and CRT display systems.
- The first stage of the program is the development of validated data and the display of that data in a useful summary to the operator. Validated data are based on the computed (statistical) analysis of redundant sensors combined with analytic process calculations giving so-called "analytic redundancy". Display of the validated data from a highly reliable system will relieve the operator of the need to rapidly decide which redundant instrument is reading correctly.
- With valid data and a system of analytic models tracking the process, the next stage is to diagnose faults. At the lowest level, the sensor failures are a natural output of the data validation. At a higher level, the equipment status and system integrity (e.g. pump failure or pipe breaks) can be monitored and rapidly brought to the attention of the operator.
- As confidence develops from the demonstrated performance and reliability of the data validation and diagnostic instrumentation, basic research is also in progress on the possible future strategies for future digital control systems. This control may initially be on a subsystem only and finally a complete closed loop control will be considered.

The first application of this conceptual design is expected to be in the area of Safety Parameter Display System (SPDS, as described in NUREG-0696). The methodology, with validated signals utilizing analytic redundance, is capable of meeting the accuracy and high reliability required for the SPDS integrated safety function information and display.

Another area of research appropriate for a university is a basic study of unified systematic approaches to the use of distributed digital processors, in a fault-tolerant array, for complete closed loop control of nuclear power reactors.

An example of part of this program is the active research study and demonstration of advanced reactor control which has been in progress at M.I.T., using the M.I.T. reactor. Figure 6.A.4 shows the basic configuration of the digital control system being demonstrated. A recent report describing this work is also enclosed in Annex C. In this work the power regulation of the M.I.T. reactor is controlled by the motion of a control rod by utilizing the validated signal from a set of redundant sensors. The control methods tested are analog and digital. The non-human control methods have been utilized in different feedback constraint modes. The results are summarized in the attached paper. It has been seen, for the application being tested, that digital control permits the most successful (and most flexible) attainment of operational goals.

Work is continuing on improved closed loop control techniques and development of the distributed processing system for use in power reactor.



LSI-1123 MICROPROCESSOR)
SYSTEM FROM DATA TRANSLATION)

Fig. 6.A.4 Digital Control Scheme for MITR-II

7. GETTING FROM HERE TO THERE

In the preceding section, several promising lines of engineering development were identified which in principle could significantly enhance the competitiveness of the nuclear option in the 1990s and thereafter. Given the present state of the industry, however, few of these ideas will be readily implemented. As noted in the introduction, the electric utilities are primarily preoccupied with their operating nuclear plants and with those that are still under construction. For most utilities, the response to the problems experienced with many of these plants has been to draw back from the nuclear option rather than to encourage a search for ways to make it more attractive in the longer run. The present generating capacity surplus has influenced this attitude, as has the general erosion of political support for nuclear power. There is also concern that a significant departure from conventional light water reactor designs, if it appears to be motivated by an effort to reduce risks, would heighten public suspicions about the safety of existing plants, and could trigger another costly round of regulatory ratchetting.

The vendors have different but at least as compelling reasons not to invest heavily in the development of advanced nuclear power plant systems. The potential market is too many years into the future, and its magnitude is uncertain. Nuclear power plant manufacturing has never been profitable in any case. And the successful commercialization of advanced nuclear plants would of course take place at the expense of existing designs, in which, moreover, the vendors continue to claim confidence. A modest-sized design improvement program is acceptable to the vendors and even desirable, if only to preserve a residual system design capability for a possible nuclear recovery; but, at least for Westinghouse and General Electric, this is being achieved through the collaborations with their Japanese licensees and partners. For the foreseeable future, the U.S. vendors will continue to concentrate on the more profitable nuclear services sector.

Finally, against a political background of rapidly growing federal budget deficits, a vocal and active political opposition to nuclear power, and sharp cutbacks elsewhere in the federal energy research and development budget, the prospects for a major government-funded nuclear power plant innovation program are also poor. The political situation is further complicated by the breeder controversy. An influential element of the natural constituency for a government-backed program of design innovations targeted for commercialization in the mid-1990s would withhold its support if the program was perceived as a threat to the breeder, and to the Clinch River Breeder Reactor in particular. Among this group, the recognition that a successful intermediate-term innovation program would ultimately enhance the prospects for breeder commercialization is apparently overshadowed by the fear that breeder opponents would characterize such a program as a viable alternative to the breeder, and that it would allow wavering supporters of the Clinch River reactor to dissociate themselves from that project without seeming to embrace an antinuclear position.

Table 7.1, which compares the current political and economic climate for technological innovation with the conditions under which the light water

Table 7.1

Comparison of the incentives for nuclear power plant innovation in the late 1950s/early 1960s and the early 1980s*

	Late 1950s/Early 1960s	Early 1980s
Utilities:	A place in the vanguard of an emerging industry important to some utilities. Public relations value high.	Prestige attached to nuclear industry leadership eroded. Public relations value low or negative.
	Utilities generally in sound financial postion. Declining marginal costs. Low interest rates.	Utility financial position much weaker.
	Fear of government ownership of electric power generation provided incentive for utility participation in reactor commercialization.	Less fear of government ownership.
	Public utility commissions generally favorable towards utilities and towards nuclear.	PUCs less sympathetic to utilities. Electricity rates increasing. Nuclear less popular. PUCs under political pressure to oppose new nuclear projects.
	Hospitable climate for safety and environmental regulation of new plant designs.	Historical record of difficulties with LWR licensing a deterrent. Also possibility of adversely affecting existing plants.
	Period of steady, rapid growth in electric power demand.	Demand growth much slower. Large uncertainties over future demand behaviour.
	Optimism over nuclear economics. Expectation of long-run decline in costs.	Historical record of unanticipated major cost escalations with LWR plants a strong deterrent.
	Attractive loss-leader turnkey plant offers by vendors.	Vendors unwilling to bear financial risks.
Vendors:	Expectation of major long-term profit opportunities. Early market penetration important. Rapid market growth seen.	Future market for nuclear plants uncertain. Vendors sustained heavy financial losses with LWR commercilization. Risks perceived to be much greater. Services market more profitable.
	Major export prospects. Opportunity for world leadership. Nuclear the next technological frontier.	International market in recession. Home markets of foreign competitors protected. Proliferation problems complicate international commerce.
	Government willing to share financial risks.	Government support less certain.
Government:	Public support for nuclear power high. Atoms for Peace program. Opportunity for world leadership.	Public unenthusiastic about nuclear power. New innovation program politically as well as financially costly. Competing priorities for government energy r.&d. support. Concern over impact on breeder program.

* Adapted from Johnson, Merrow, Baer, and Alexander (1976), op. cit.

reactor emerged as a commercial option two decades ago, underlines the difficulties facing a major initiative of any kind at this stage.

How might the present situation be changed? What would be needed to stimulate more vigorous efforts to develop innovative nuclear power plant system designs for the 1990s? There are no easy institutional solutions. The obstacles are deep-rooted, and without a basic change in assumptions and policies no amount of creativity in organizational restructuring will suffice. In particular, two fundamental changes in strategy would seem to be necessary:

(i) Notwithstanding its highly fragmented structure, the electric utility industry must recognize its own pre-eminent role in ensuring the timely availability of the technology which will enable it to sustain and improve the quality of its service. Specifically, the electric utilities must focus now on the question of whether nuclear power might have an important role to play in meeting the nation's demand for new and replacement sources of electricity a decade or more in the future. If this seems likely -- and we expect this to be the case -- a group of utilities should take the lead in establishing a set of general technical specifications to be met by nuclear power plants at that time. In the longer run, the utility industry (and, ultimately, the ratepayers) must bear a substantial portion of the costs and the financial risks involved in developing the technology to meet these requirements.

(ii) Provided the private sector is demonstrably prepared to take the lead, the Federal government should strengthen the medium term component of its own portfolio of nuclear power plant research and development. At present, Federal nuclear r.&d. is primarily oriented towards the short term (in support of LWR plants in operation or under construction) and the much longer term (in support of the breeder). Federal participation along the suggested new lines should be in close support of clearly-defined electric utility objectives.*

We note that these requirements differ from the arrangements in effect during the period of LWR commercialization, and also from those pertaining today to the breeder program. In particular, they imply a significant shift in both the burden of developmental financial risk and the focus of technological decision-making towards the utilities. These shifts reflect current political and economic realities: it is evident that neither the traditional suppliers of nuclear plants nor the federal government will act in the absence of a clear commitment by the electric utilities. But they also incorporate at least two important lessons learned from previous efforts to develop and commercialize large-scale new technologies. First, greater risk and cost sharing by the utilities relative to the government is likely to enhance the quality of the information generated by the innovation

* We do not attempt to prescribe an 'optimal' allocation of risks and costs between ratepayers, utility investors, taxpayers, and plant suppliers. The issues are complex, and the political and regulatory processes for resolving them untidy. The point remains, however, that unless the electric utility industry is prepared to take the initiative no significant advance is likely.

program. Private companies are generally better judges of commercial prospects than is the Federal government, and their participation as risk-bearers is an effective filtering device to identify technologies with the greatest near-term commercial potential. Private cost and risk-sharing also means that more realistic program goals are likely to be set and that the incentives of the private participants to meet these goals will be greater.*

Second, where the success of an innovation depends upon the technological capabilities of the user as well as the supplier, as is obviously the case with nuclear power plants, the importance of a well-informed user and of good technical communication between user and supplier is self-evident. In retrospect, the lack of experience and sophistication of the electric utilities in the field of LWR technology (with

* L.L. Johnson, E.W. Merrow, W.S. Baer, and A.J. Alexander, Alternative Institutional Arrangements for Developing and Commercializing Breeder Reactor Technology, The Rand Corporation, Santa Monica, California, R-2069-NSF, November 1976. For a more recent affirmation of these arguments, see R.R. Nelson and R.N. Langlois, "Industrial innovation policy: Lessons from American history," Science, Vol. 219, 18 February 1983, 814-818. According to this historical study, of the various types of government support for civilian research and development which have been tried in the past, programs in which federal agencies have "attempted to insert themselves directly into the business of developing particular technologies for a commercial market in which they had little or no procurement interest" have enjoyed least success. The authors contrast the 'unequivocally negative' results of such programs with other types of government support in which the experience has been at least partly positive, including: programs in industries in which the government was heavily involved as a user of the technology; programs to support 'generic' technologies in the area between basic research and applied r.&d.; and programs to support applied r.&d. pursuant to well-defined client demands.

some notable exceptions) has been a major cause of the difficulties encountered in nuclear power plant construction and operation. There is also reason to believe that the utilities' limited ability to define their requirements and to articulate them to the plant designers may also have contributed to subsequent problems (although this point is inherently more difficult to substantiate). For all of these reasons, without the active technical participation of the utilities in the formulation of design specifications and, where applicable, in the construction and operation of prototypes, future nuclear power plant innovation efforts appear much less likely to succeed.

We do not attempt to prescribe fully the organizational structures needed to bring the innovations in Section 6 to commercial fruition. (How many utilities would be involved? How much would each participant contribute to the expected cost of the program? How would the financial risks of cost overruns, delays, and other unanticipated developments be distributed among the individual participants? Who would be responsible for the various technical functions of the program, including, where appropriate, development, design, construction and operation? Who would have overall management responsibility?) Such structures tend to emerge in an *ad hoc* manner, the product of the possible rather than the result of a systematic analysis of alternatives. In any case, a full discussion of these programmatic issues would not be particularly meaningful until committed 'champions' of specific innovation concepts could be identified. Nevertheless, in the following section we propose some organizational arrangements which would permit at least the next step to be taken in the exploration of some of the most promising of these innovations.

8. IMPLEMENTATION: THE ROLE OF M.I.T.

Thus far we have developed a general argument for pursuing power plant design innovations as part of a larger effort to restore the competitiveness of the nuclear option. Several promising design concepts or lines of development have been described. We now propose, in outline form, a program of research in this area to be undertaken at MIT. The program is put forward as a response to the needs and opportunities for technological innovation already identified. At the same time it reflects the capabilities, objectives and constraints of a university research effort. The program would be centered in the Nuclear Engineering Department, where many of the relevant research capabilities are located; however, it has been conceived from the beginning as a multidisciplinary effort, and would draw on the full range of faculty research interests and experience at the Institute.

We stress at the outset that the proposed program would be conducted in parallel with substantial and continuing departmental research efforts focused on longer-term options, i.e. the breeder and fusion, and on improving various aspects of the performance of the 120 or so LWRs currently in operation or under construction. These latter plants will continue to demand most of the attention of the nuclear utilities and their suppliers, and our own research profile will reflect this emphasis.

A very large commitment of technical, financial, and probably also political resources will be required to bring about major innovations in nuclear power plant technology. No university research effort can make more than a minor contribution on this scale. In the shorter run, however, the role of an independent research group such as exists at M.I.T. may be more important. As the previous discussion has implied, in the present difficult climate for nuclear power the major industrial participants can point to many good reasons not to take initiatives in this area, even while acknowledging that more attention to it could yield generally beneficial results. To this extent, a small, independent research group which is prepared to address constructively some of the more difficult, speculative or sensitive design-related issues may have an important role in catalyzing the much larger-scale effort that would be required for full commercial implementation.

The proposed research program will center on two large-scale engineering development projects. It will also include supporting engineering and social science research projects and policy studies, as appropriate.

8.1 Development of specific plant design innovations

We have selected two general areas of emphasis for engineering development: evolutionary innovations in LWR designs; and small, modular high temperature gas reactor systems. Expanding the list any further would lead to an unacceptable dilution of faculty effort. The two areas chosen reflect current faculty research interests as well as a general assessment of relevance to the national need.

8.1.1 LWR Innovation

It is proposed to carry out a research project in association with a consortium of leading electric utilities with the objective of developing a design specification for an advanced LWR responsive to the projected needs of utilities in the 1990s. Specifically, the project is intended to achieve the following:

- The definition of a comprehensive set of design goals for an optimized LWR;
- Investigation of the technical requirements implied by these goals at least to the extent of demonstrating that a feasible design solution exists; and
- Use of these goals as the basis of a specification for subsequent LWR plant construction projects to be undertaken by participating utilities. The actual design specification of a new plant would not be formulated in this project. Rather, this step would be undertaken by a participating utility, possibly with the participation of the project staff. However, the basis of that specification would be established in this project.

The consortium assembled for this project would ideally consist of between four and eight large technically-advanced utility companies and a team to be assembled from MIT faculty and staff. Each utility would assign two staff members to work on the project full-time, either at the utility site or at MIT, depending upon the nature of the work to be performed.

Overall responsbilities for the project would rest with the leader of the MIT team. The immediate task of the group would be to formulate a comprehensive inventory of the design goals (and their justifications) which should be incorporated in the companies' next LWR project, and to rank them in importance. It is expected that these lists of design goals would vary from one utility to another but that they would have a sufficient amount in common to permit the development of a consensus regarding the highest priority areas of design specification. These in turn would probably form the central focus of the project. As discussed in Section 6, some probable examples of these areas of focus include the following:

- Reliability improvements.
- Simplification.
- Refinement of plant safety criteria.
- Unit size optimization.
- Minimization of maintenance and equipment replacement costs and radiation dose burdens.

Many of these areas of potential design change are expected to require significant refinement before a practical, unambiguous and optimal design specification can be expressed, and with each such specification it will be essential to demonstrate that at least one practical design approach exists by which it can be met. These tasks of refinement and demonstration of practicality are expected to consume most of the resources of the project.

The intended project duration is five years. With a project of this sort there is no natural duration, in that as higher priority design issues are resolved attention would shift to other outstanding topics. The current list of such issues is long enough that this effort could occupy several tens of man-years. The five year project duration is selected because this is a period sufficient to make substantial progress on a large number of the highest priority topics, and also because it is compatible with the planning window of many utilities with regard to new baseload capacity additions.

PWR vs BWR: Because of the many significant differences between PWR and BWR technologies and because of the preponderance of PWRs in United States and world nuclear power programs, it is intended that the initial and primary focus of the project would be upon the PWR type. In such a project it would remain possible to address a limited, but important, set of BWR innovations in areas where the technology is similar (eg. materials corrosion and water chemistry, pressure vessel neutron embrittlement, information system and robotic design). Further, it would be possible to undertake a separate BWR effort if sufficient interest were to exist among project participants, or once the project were fully underway on the PWR phase.

It is anticipated that the supporting investigations will be performed equally by the utility and MIT project participants. This structure is needed to ensure the continued dynamic involvement of the utilities and the MIT participants in the development of the design goals and to avoid segregation of the consortium into a group of analysts and a group of their sponsors. The participation of experienced and knowledgeable utility personnel is essential to the success of this project.

The work of the consortium is thus envisioned to consist of a group of continuing projects which would contribute to the resolution of questions arising in the iterative refinement of the design goals, with periodic meetings, workshops and other communications for review of findings, project redirection and design goal clarification. It is expected that the MIT researchers involved in this work would function in a fashion similar to that of the utility participants, contributing as a group to the continuing work of design goal definition, and individually conducting research -- typically working with graduate student research assistants -- needed to advance the understanding of the various possible LWR design improvements. Examples of current investigations at MIT which would be relevant to the project are presented in Section 6.

Budget and Timescale

The proposed project would be completed in approximately five years. A consortium membership of between four and eight utilities would be most desirable in order for the proposed work to make a substantial contribution. Each utility would be asked to contribute approximately $200,000 per year to support the work of two MIT researchers and two graduate research assistants and the necessary administrative functions (with adjustments for inflation). The anticipated MIT project budget would thus be approximately $0.8-1.6million/yr. In addition, each utility would be asked to assign two staff members to work full time on design goal definition and refinement, and to arrange for a Vice President to manage the utility's interactions with the project. A total of 24-48 technical staff (including 8-16 graduate students) would thus be involved.

8.1.2 Small modular HTGR systems studies

In Section 6 several research projects were suggested in connection with a proposed new MIT initiative in the area of small, modular HTGR systems. Our ultimate goal is to determine whether small, modular HTGRs should be promoted as next generation power systems, and, if so, how the development program should be structured. The initial stages of the program are aimed at defining more fully the properties of this class of systems. They will include:

- Comparison studies of prismatic versus pebble-bed modular HTGR designs including both safety and economic considerations.
- Core physics and neutron economy in very small reactors.
- Studies of computer control of the modular system including possible applications of robotics to maintenance.
- Studies of the containment system requirements and designs for the modular arrangement.
- Assessment and study of improved modular steam generator designs.

Although we believe that this technology offers considerable promise over the time frame of interest, developing the commitment necessary to move it towards commercial demonstration in an industry which historically has not shown much enthusiasm for gas reactor systems will obviously be a formidable task. Evidently the approach here will be much different from that proposed in the LWR case. We envisage an incremental process, with attempts made at each stage to increase the level and scope of participation. In view of the narrow base of support for gas reactor systems in the U.S., we would seek to build international links from the outset. In the United States, Gas Cooled Reactor Associates (GCRA) is probably the key organization. GCRA is a utility financed group established some years ago to evaluate and plan the commercial deployment of gas cooled reactors. At present, some 29 utilities contribute approximately $2 million annually to GCRA. Most support for gas reactor research and development has been provided until now by the Department of Energy, but only a small fraction of this has been assigned to small modular reactor studies. This small funding is supporting work in progress at General Atomic, General Electric and Bechtel. The studies involve both prismatic and pebble bed modular systems. (GE is not supporting this effort with any internal funds.) Overseas, KFA Juelich in West Germany is pursuing small modular systems most actively, as discussed in Section 6. Contacts with members of all of these organizations have been established.

The next step towards obtaining funding for the proposed studies at MIT will be to review the program with the GCRA management. Early contacts indicate that they will be willing to encourage funding of our work. We anticipate next approaching interested GCRA sponsors for research support. If some funding is forthcoming, the fact of utility interest could be used in support of a subsequent request for government funding. Support from the

Department of Energy will depend on Congressional actions with regard to the DOE HTGR budget. Our potential for funding may thus be enhanced if the interested utilities are willing to provide additional signals to Congress.

Budget and Timescale

The program at MIT is envisaged to start with at least two interested professors and build up to 4 or 5 supported students per professor. The target funding level for the project is $0.5 - 1million/yr. The first phase of the project will consist of a comprehensive assessment of the acceptability of the small, modular HTGR concept. It will be completed in 3-5 years, depending on the rate at which the target funding level is approached.

8.2 Supporting engineering and social science research and policy studies

8.2.1 Supporting Research.

During the course of this study, several research topics have been identified which, while not specific to one design concept or another, would nevertheless strengthen a general program of research on nuclear power plant innovations, while at the same time providing excellent opportunities for student thesis research projects. Some of these topics conceivably could be supported under the technical projects described previously, but in other cases separate funding will be necessary. Examples include:

- Further research on the relationship between technological hazard characteristics and the risk perceptions of the general public, and the implications of this relationship for power plant design goals;
- Research on the risk perceptions of utility investors and managers, and the implications for future innovation efforts;
- A comparative study of power plant construction experience across a sample of U.S. utilities in order to identify and quantify the primary contributors to the variance in construction lead-time and plant capital cost;
- A comparative analysis of specific industrial innovation decisions and programs in the nuclear power plant and civilian aircraft industries in order to investigate the impacts of industry structure and organization on the process of large-scale technological innovation:
- An economic analysis of the optimum baseload unit size as a function of demand growth, cost of capital, supply system size, generating mix, utility power pooling, construction lead times, and scale economies.

8.2.2 Policy Studies.

It is also proposed to carry out policy-related studies which would contribute to the development of a framework for evaluating alternative advanced nuclear power plant options and innovation strategies at the national level. Such studies would include the technical review of a range of potentially promising design innovations (including but obviously not limited to those under active investigation at MIT): the identification of the key engineering steps required to bring each to the point of trial use; and the assessment of the organizational, financial and regulatory demands associated with these innovations.

8.2.3 Funding.

Possible sources of funding for the research proposed in this category include the Department of Energy, the National Science Foundation and several private foundations.

A research grant of $176,000 has been received from the National Science Foundation to undertake a preliminary assessment of alternative national nuclear power plant innovation strategies during an 18-month period beginning in September 1983. The NSF-sponsored project will involve four faculty members and three graduate students. The faculty members will also be participating in other areas of the overall innovation program.

ANNEX A

ANALYSIS OF LOSS OF FEED WATER ATWS TRANSIENTS

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ABSTRACT

Analysis of Anticipated Transients without Scram (ATWS) forms an important part of the assessment of the safety of Pressurized Water Reactors (PWRs). The loss of feed water anticipated transient, in which the secondary side of the steam generators suffers a boil-off, leads to high temperatures and high pressures in the primary system. The prediction of the course of this ATWS transient is the subject of this paper. A simple computer code based on lumped parameter modeling has been developed to predict the transient pressures and temperatures. The results compare very well with those obtained from the RETRAN code. The computer run time is approximately 1/15th of the transient real time.

I. INTRODUCTION

The loss of feedwater (LOFW) transient in a PWR degrades the heat removal from the primary water due to the boil-off of the water on the secondary side of the steam generators. This condition, if not accompanied by a scram of the reactor power (ATWS) will lead to temperature increase and pressurization of the primary side. The opening of safety valves discharges flow and energy from the reactor system and the reactor power decreases due to the negative reactivity feedback from the increase in the moderator temperature. The reactor pressure eventually decreases due to the coolant loss and the power decrease. High values of pressure, (above critical pressure), however, may be reached during this transient for a typical PWR.

Analysis of loss of feedwater anticipated transients without scram (LOFW ATWS) are usually performed with the thermal hydraulic systems codes such as RETRAN (1). RETRAN has the capability of modeling a wide variety of transients in both pressurized and boiling water reactors, but suffers from long running times. This paper describes a simpler fast-running computer code for modeling this transient. Typical running times obtained have been approximately 1/15th the transient real time.

The code is presently restricted to the LOFW ATWS in PWRs with U-tube steam generators, but can be extended to a wider class of PWR transients. For instance, the LOFW transient with reactor scram can be treated easily with the addition of a scram reactivity curve and a decay heat curve. The addition of a momentum equation for the primary loop would allow modeling of a pump coastdown. The code should be able to model the loss of load, overcooling, and steam line break transients with minor modifications of the steam generator model.

II. OVERVIEW OF THE CODE

The system model includes representations of the core, primary coolant loop, pressurizer, and the secondary system. Figures 1 and 2 illustrate major aspects of the plant representation. The core is represented by an average coolant channel coupled to a fuel pin model. Changes in reactor power due to fuel and moderator temperature feedbacks are simulated using the neutron point kinetics equations. The primary coolant loop is represented by several volumes or nodes, as illustrated in Figure 1. The major assumptions concerning the primary system are: (1) pressure drops between components are negligible in comparison to the system pressure; (2) the coolant loops operate symmetrically, thus allowing their represenation by a single lumped loop; and (3) the volumetric flow rate of the primary coolant is constant in time, i.e, the pumps remain in operation. The pressurizer is modeled as a two region nonequilibrium volume with sprays and relief valves. The secondary system model (Figure 2) consists of a steam generator, main steam and bypass lines, valves, and a constant pressure volume representing the condenser. A detailed description of these models is given in the next section.

Figure 3 is a flow chart illustrating the major subroutine calls of the main program. Subroutine READIN reads input data and performs a steady-state initialization. The steady-state initialization

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minimizes the amount of input required and insures a consistent set of initial conditions. After initialization, a loop through time is begun. Subroutine PKNTCS is called first each time step. Input to this subroutine includes the average temperatures of the fuel and coolant and output is core power. Subroutine TEMP solves the energy conservation equations for the fuel pin and primary coolant nodes. The output of this subroutine are fuel and coolant temperatures. Subroutine PZR determines the primary system pressure by solving the conservation of mass and energy equations for the primary system and pressurizer simultaneously. Input to this subroutine is the net heat input to the primary coolant, which is computed in subroutine TEMP. Subroutine STMGEN computes steam generator secondary side temperature and mixture level. These parameters are used by subroutine TEMP to determine the heat transfer rate from primary to secondary coolant during the next time interval.

III. PRIMARY SYSTEM ENERGY EQUATIONS

This section describes the formulation and solution technique of the primary system energy equations. The primary system as referred to here includes the fuel pin and primary coolant loop, excluding the pressurizer. These equations are used to compute average temperatures in the fuel and coolant.

A. Fuel Pin Heat Conduction

The fuel pin heat conduction equation is solved by dividing the fuel pin into several radial nodes and using a lumped parameter equation to simulate each node. Up to ten radial nodes may be used to simulate the fuel pin. The following lumped parameter equation is used to represent each node:

$$m_i c_i \frac{dT_i}{dt} = q_i'''V_i \frac{T_{i-1} - T_i}{R_{i-1}} - \frac{T_i - T_{i+1}}{R_i} \qquad (1)$$

where:

m_i = mass of node i,
c_i = specific heat of node i,
T_i = average temperature of node i,
q_i''' = volumetric heat generation rate in node i,
V_i = volume of node i, and
R_i = thermal resistance between node i and node i+1

B. Primary System Temperature

Figure 1 illustrates the nodalization of the primary coolant loop. The core is represented by a single average coolant channel. The steam generator U-tube region is represented by two coolant nodes and two wall nodes. Nodes are also provided for a lumped lower plenum and cold leg and a lumped upper plenum and hot leg.

An energy balance on any coolant node j, neglecting changes in potential and kinetic energies, yields:

$$\frac{d}{dt}(m_j h_j) = \dot{m}(h_{ji} - h_{jo}) + \dot{Q}_{inj} - \dot{Q}_{outj} + CV_j \frac{dP}{dt} - \dot{m}_{outj} h_j \qquad (2)$$

where

m_j = coolant mass in node j,
h_j = average enthalpy,
$\dot{m}$ = mass flow rate of primary coolant
h_{ji} = enthalpy of liquid entering node j,
h_{jo} = enthalpy of liquid exiting node j,
$\dot{Q}_{inj}$ = total heat input to node j,
$\dot{Q}_{outj}$ = total heat extracted from node j,
V_j = volume of node j,
C = conversion factor
$\dot{m}_{outj}$ = mass flow out of node j due to volume expansion of the primary coolant

Note that it was implicity assumed that the mass flow rate of the primary coolant is uniform throughout the system. The term $\dot{m}_{outj}$ represents the mass decrease due to the expansion of the primary coolant. It is included only for completeness and is eliminated below. The mass conservation equation for node j is simply:

$$\frac{dm_j}{dt} = \dot{m}_{outj}$$

Assuming there is no phase change in the primary coolant, temperature may be related to enthalpy through the state equation by:

$$dT_j = \left.\frac{\partial T}{\partial h}\right)_P dh_j + \left.\frac{\partial T}{\partial P}\right)_h dP \qquad (3)$$

Defining the specific heat as $C_P = (\partial h/\partial T)_P$ and combining equations (1), (2) and (3) yields the following equation for temperature:

$$\frac{dT_j}{dt} = \frac{\dot{m}\, C_P(T_{ji} - T_{jo}) + \dot{Q}_{inj} - \dot{Q}_{outj}}{m_{jPj}} + \left[\frac{CV_j}{m_j C_{Pj}} + \left.\frac{\partial T}{\partial P}\right)_h \right] \frac{dP}{dt} \qquad (4)$$

Equation (4) is invalid if there is a phase change (boiling) in the node. If the temperature of any node reaches the saturation temperature, execution is halted and a message printed.

Equation (4) is simplified for various nodes as follows. For the upper and lower plena, perfect mixing is assumed. That is, the temperatures of the coolant exiting the plena equals the average coolant temperatures of the plena ($T_{jo} = T_j$). The average temperature of all other nodes is related to the inlet and outlet temperatures by:

$$T_j = \frac{1}{2}(T_{ji} + T_{jo}) \qquad (5)$$

For the core node (j = 1), $\dot{Q}_{out} = 0$ and

$$\dot{Q}_{in} = \frac{T_c - T_1}{R_c} \qquad (6)$$

where T_c is the average clad temperature and R_c is the thermal resistance through the clad plus the surface resistance. Equation (6) couples the fuel pin equations to the coolant energy equations.

For the steam generator nodes, $\dot{Q}_{inj} = 0$ and

$$\dot{Q}_{outj} = \frac{T_j - T_{j,wall}}{R_j} \qquad (7)$$

where $T_{j,wall}$ is the U-tube wall temperature and R_j is a thermal resistance. The steam generator U-tubes are represented by two nodes, one each for the hot and cold sides.

C. Core Power

Core power will change in time due to the temperature feedback effects from the moderator and fuel. To model this effect, the point kinetics equations with a prompt-jump approximation are used. The delayed neutron precursors are represented by two effective groups. Reactivity is calculated by:

$$\rho_{FB} = \alpha_F(\bar{T}_F - T_{Fo}) + \alpha_m(\bar{T}_m - T_{mo}) \tag{8}$$

where

$\rho_{FB}(t)$ = reactivity due to temperature feedbacks
α_F = fuel temperature coefficient of reactivity
$\bar{T}_F(t)$ = average fuel temperature
T_{Fo} = reference fuel temperature
$\bar{T}_m(t)$ = average moderator temperature,
T_{mo} = reference moderator temperature.

To permit an analytical solution of the kinetics equations, the reactivity is considered constant over each time step.

D. Solution Technique

The fuel pin heat conduction equations (1) and primary coolant energy equations (4) form a coupled set of first order differential equations. The equation set may be represented by:

$$\frac{d\bar{T}}{dt} = \bar{\bar{A}}\,\bar{T} + \bar{q}$$

The vector $\bar{T}$ contains the temperatures of fuel, clad, primary coolant and steam generator U-tubes. The coefficient matrix $\bar{\bar{A}}$ contains thermal inertia and resistance terms. The vector $\bar{q}$ contains heat sources terms (from fission), heat sink terms (steam generator), and a term to account for changes in temperature due to pressure change (see Equation 4). The equation system is linearized by assuming the coefficients of $\bar{\bar{A}}$ and $\bar{q}$ are constant over each time step. The system is integrated over each time step by the trapezoidal rule, a second order method which is stable for any time step size.

IV. PRIMARY SYSTEM PRESSURE

The primary system pressure is calculated by combining mass and energy balances for the primary coolant loop with mass and energy balances for the pressurizer and the equation of state. The pressurizer is modeled as a two region volume, one region being liquid and the other predominantly vapor. The two regions are assumed to be at the same pressure but not necessarily the same temperature. The rest of the primary system is modeled as a single lumped volume.

A. Pressurizer

The conservation of mass and energy equations for the liquid region may be written concisely as:

$$\frac{dM_L}{dt} = \sum_{jL} W_{jL} \tag{9}$$

and

$$\frac{d}{dt}(M_L h_L) = \sum_{jL} W_{jL}\, h_j + CV_L \frac{dP}{dt} \tag{10}$$

where

M_L = mass of liquid region
h_L = enthalpy of liquid region,
W_{jL} = mass flow rate into or out of liquid region,
h_{jL} = enthalpy of mass entering or leaving liquid region,
V_L = volume of liquid region,
C = conversion factor
P = pressure
Σ = denotes the summation over all flow paths j into or out of the region.

Equations (9) and (10) can be combined to give:

$$\frac{dh_L}{dt} = \frac{\sum_{jL} W_{jL}(h_{jL} - h_L)}{M_L} + Cv_L \frac{dP}{dt} \tag{11}$$

where v_L is the specific volume of the liquid region.

For the vapor region, there are no heat sources and equation (11) takes the form:

$$\frac{dh_v}{dt} = \frac{\sum_{jv} W_{jv}(h_{jv} - h_v)}{M_v} + Cv_v \frac{dP}{dt} \tag{12}$$

where

M_v = mass of vapor region
h_v = enthalpy of vapor region
W_{jv} = mass flow rate into or out of vapor region
h_{jv} = enthalpy of mass entering or leaving vapor region,
v_v = specific volume of vapor region,
Σ = denotes the summation over all flow paths into or out of the vapor region.

The change in volume for each region may be written in terms of changes in enthalpy, pressure and mass. For the liquid region,

$$\frac{dV_L}{dt} = v_L \frac{dM_L}{dt} + M_L \frac{dv_L}{dt} \tag{13}$$

Introducing equation (9) into (13) yields

$$\frac{dV_L}{dt} = v_L \sum_{jL} W_{jL} + M_L \frac{\partial v_L}{\partial t} \tag{14}$$

From the equation of state we may write:

$$\frac{dv_L}{dt} = \left.\frac{\partial v_L}{\partial h_L}\right)_P \frac{dh_L}{dt} + \left.\frac{\partial v_L}{\partial P}\right)_{h_L} \frac{dP}{dt} \tag{15}$$

Introducing this into equation (14) we obtain:

$$\frac{dV_L}{dt} = v_L \sum_{jL} W_{jL} + M_L\left[\left.\frac{\partial v_L}{\partial h_L}\right)_P \frac{dh_L}{dt} + \left.\frac{\partial v_L}{\partial P}\right)_{h_L} \frac{dP}{dt}\right] \tag{16}$$

Similarly, for the vapor region:

$$\frac{dV_v}{dt} = v_v \sum_{jv} W_{jv} + M_v \left[\frac{\partial v_v}{\partial h_v}\right)_P \frac{dh_v}{dt} + \frac{\partial v_v}{\partial P}\bigg)_{h_v} \frac{dP}{dt} \bigg] \quad (17)$$

The total volume of the pressurizer is constant. This implies that:

$$\frac{dV_L}{dt} + \frac{dV_v}{dt} = 0 \quad (18)$$

Equation (16) and (17) can now be combined and Equations (11) and (12) can be used to eliminate the unknown enthalpies. The result is a single equation for pressure:

$$\frac{dP}{dt} = -\left\{ M_L \left[Cv_L \frac{\partial v_L}{\partial h_L}\right)_P + \frac{\partial v_L}{\partial P}\bigg)_h \bigg] + M_v \left[Cv_v \frac{\partial v_v}{\partial h_v}\right)_P + \frac{\partial v_v}{\partial P}\bigg)_{h_v} \bigg] \right\}^{-1} \times \left\{ v_L \sum_{jL} W_{jL} + \frac{\partial v_L}{\partial h_L}\bigg)_P \left[\sum_{jL} W_{jL}(h_{jL}-h_L) \right] + v_v \sum_{jv} W_{jv} + \frac{\partial v_v}{\partial h_v}\bigg)_P \left[\sum_{jv} W_{jv}(h_{jv}-h_v) \right] \right\} \quad (19)$$

The terms $\sum_{jL} W_{jL}$, $\sum_{jv} W_{jv}$, etc., are expanded below:

$$\sum_{jL} W_{jL} = W + W_{cd} + W_{rain} - W_{flash}$$

$$\sum_{jL} W_{jL}(h_j - h_L) = W_{srg}(h_{srg} - h_L) + (W_{sp} + W_{cd} + W_{rain})(h_f - h_L) - W_{flash}\ (h_g - h_L),$$

$$\sum_{jv} W_{jv} = W_{boil} - W_{cd} - W_{rain} - W_{Rv} - W_{sv},$$

$$\sum_{jv} W_{jv}(h_{jv} - h_v) = W_{boil}\ (h_g - h_v) - W_{rain}\ (h_f - h_v),$$

where

W_{srg} = inlet surge flow rate
W_{sp} = flow rate of spray,
W_{cd} = condensation rate of vapor on spray,
W_{rain} = rainout of liquid droplets from vapor region,
W_{flash} = vapor generation rate in liquid region,
W_{Rv} = mass flow rate through the relief valve(s),
W_{sv} = mass flow rate through the safety valve(s),
h_{srg} = enthalpy of surge flow,
h_f = saturation enthalpy of liquid phase
h_g = saturation enthalpy of vapor phase.

The spray flow rate is computed from a user defined linear function of pressure. Auxiliary models are employed to compute the vapor condensation rate, rainout and flashing flow rates, and the mass flow rates through relief and safety valves. These models are described below.

Condensation

The condensation rate on spray droplets is calculated by assuming the condensate comes into equilibrium with the spray. With this assumption, an energy balance yields

$$W_{cd} = \frac{h_f - h_{sp}}{h_v - h_f}\ W_{sp} \quad (20)$$

Rainout

The code does not allow subcooled vapor states, and any liquid condensing in the vapor region is assumed to deposit instantly on the liquid surface. In order to implement this assumptions, the rainout flow rate is calculated from:

$$W_{rain} = \frac{x_v M_v}{\Delta t} \quad (21)$$

where

$$x_v = \frac{h_v - h_g}{h_g - h_f} \quad (22)$$

where Δt is the time step size.

Flashing

The flashing term is the liquid region equivalent of the rainout term. Superheated liquid states are not allowed, and any vapor formed in the liquid region is assumed to instantly appear in the vapor region. Hence, the flashing flow rate is calculated from:

$$W_{flash} = \frac{x_L M_L}{\Delta t} \quad (23)$$

where

$$x_L = \frac{h_L - h_f}{h_g - h_f} \quad (24)$$

Choked Flow

The homogeneous equilibrium model is used to compute the critical mass flux through the relief and safety valves. The critical mass flux is then multiplied by user supplied valve flow areas and conservatism factors to obtain the mass flow rate.

The unknowns in equation (19) which are not supplied as input or computed from auxiliary models are P and the surge line flow rate W_{srg}. Hence another equation is needed to close the set.

B. Surge Line Flow Rate

The surge line connects the pressurizer with the hot leg of the primary coolant loop. The surge line flow rate is computed by assuming that pressure changes in the coolant loop and pressurizer are equal. Combining mass and energy balances on the coolant loop with the equation of state results in the following equation for pressure:

$$\frac{dP}{dt} = \frac{(W_{srg} + W_{sp}) v_p \left.\frac{\partial v_p}{\partial h_p}\right)_P \left[W_{srg}(h_{srg} - h_p) + W_{sp}(h_{sp} - h_p) - \Delta\dot{Q} \right]}{M_p \left[c_p \left.\frac{\partial v_p}{\partial h_p}\right)_P + \left.\frac{\partial v_p}{\partial P}\right)_{h_p} \right]} \quad (25)$$

where
- $\Delta\dot{Q}$ = net heat input rate to primary system,
- v_p = specific volume of primary coolant,
- h_p = average enthalpy of primary coolant,
- M_p = mass of primary coolant.

An expression for the surge line flow rate may now be found by equating the right hand sides of equations (19) and (25) and solving for W_{surg}.

V. SECONDARY SYSTEM

The secondary system model is illustrated in Figure 2. The model consists of a three region steam generator, steam lines, valves, and a condenser volume. The purpose of the model is to calculate heat transfer conditions, i.e., temperature and heat transfer coefficients, on the secondary side. Therefore, some phenomenon (such as recirculation flow) which do not strongly affect these parameters are not modeled.

A. Pressure

The steam generator pressure is calculated by assuming the secondary fluid is an equilibrium mixture. The conservation of mass and energy equations for the entire steam generator, neglecting potential and kinetic energy terms, are:

$$\frac{dU_{sg}}{dt} = \dot{Q}_{sg} + (W_{in} h_{in} - W_s h_g) \quad (26)$$

$$\frac{dM_{sg}}{dt} = (W_{in} - W_s)\ dt \quad (27)$$

where
- M_{sg} = total water mass in steam generator,
- U_{sg} = total internal energy,
- $\dot{Q}_{sg}$ = heat transfered to steam generator
- W_{in} = inlet flow rate to steam generator
- h_{in} = enthalpy of inlet flow,
- W_s = flow rate of steam exiting steam generator, and
- h_g = enthalpy of steam.

Once internal energy and steam generator water mass are known, pressure may be calculated from the equation of state.

B. Mixture Level

The two phase mixture level in the U-tube bundle region is computed from:

$$z_{mix} = \frac{z_{bc}}{(1 - \bar{\alpha})} \quad (28)$$

where
- z_{mix} ≡ two phase mixture level,
- z_{bc} ≡ collapsed liquid level, and
- $\bar{\alpha}$ ≡ average void fraction in the bundle region.

The collapsed liquid level is calculated from a mass balance on the steam generator. It is assumed that the collapsed liquid levels (static head) in the downcomer and U-tube region are equal. The collapsed liquid level may then be computed from:

$$z_{bc} = \frac{g_c}{g} \frac{M_\ell}{(A_b - A_d)\ \rho_f} \quad (29)$$

where
- g_c = conversion factor
- g = gravitational constant,
- M_ℓ = total liquid inventory
- A_b = flow area of bundle region,
- A_d = flow area of downcomer, and
- ρ_f = density of liquid.

The average void fraction is defined by;

$$\bar{\alpha} = \frac{1}{z_{mix}} \int_0^{z_{mix}} \alpha(z)\ dz \quad (30)$$

The average void fraction is computed numerically by dividing the boiling region into axial steps. In this calculation, the mixture level at the previous time step is used.

The void fraction at each axial step is computed from either the drift flux equations (2) or the empirical Yeh (3) correlation.

The drift flux model expresses the local void fraction as:

$$\alpha = \frac{j_g}{C_o(j_g + j_f) + V_{gj}}, \quad (31)$$

The parameter C_o is a distribution parameter which arises from radially averaging the void fraction and fluid velocities. It typically ranges from 1.0 for bubbly flow to around 1.2 for fully developed slug flow. The quantity v_{gj} is the vapor drift velocity. It is computed from:

$$V_{gh} = 1.41 \left[\frac{\sigma g g_c (\rho_f - \rho_g)}{\rho_f^2} \right]^{1/4}, \quad (32)$$

where σ is the surface tension and j_g and j_f the superficial velocities are computed from:

$$j_g(z) = \frac{\dot{Q}_{sg} - W_{bi}\ \Delta h_f}{\rho_g A_b L_b \cdot h_{fg}}\ z \quad (33)$$

$$j_f(z) = \frac{W_{bi}}{\rho_f A_b} - \frac{\rho_g}{\rho_f} j_g(z) \quad (34)$$

where
- W_{bi} = flow rate from the downcomer to the bundle,
- Δh_f = subcooling of liquid in downcomer,
- L_b = height of U-tube bundle, and
- h_{fg} = latent heat of vaporization.

An equation for the flow rate W_{bi} may be derived by writing mass balance equations for the bundle region and downcomer and equating the collapsed levels. The result is

$$W_{bi} = \frac{A_b W_{in} + A_d \dot{Q}_{sq}/h_{fg}}{A_b + A_d(1 + \Delta h_f/h_{fg})} \tag{35}$$

where W_{in} is the feedwater (or auxiliary feedwater) flow rate.

The Yeh correlation expresses the local void fraction as:

$$\alpha = 0.925 \left(\frac{\rho_g}{\rho_f}\right)^{0.239} \left(\frac{j_g}{V_{bcr}}\right)^{b} \left(\frac{j_g}{j_g + j_f}\right)^{0.6} \tag{36}$$

where the critical superficial steam velocity V_{brc} is given by:

$$V_{bcr} = \sqrt{\frac{2}{3}} \left[g \left(\frac{1.53}{2/3}\right)^2 \left(\frac{\sigma}{g\rho_f}\right)^{1/2} \right]^{1/2} \tag{37}$$

and the power coefficent in Eq. (36),

$b = 0.67$ if $j_g/V_{brc} < 1$
$b = 0.47$ if $j_g/V_{bcr} > 1$

Both the Yeh correlation and the drift flux model were employed in determining the two phase mixture level as a function of time in the FLECHT-SEASET (4) and the Westinghouse 336 rod bundle (5) boil-off experiments. Examples of the comparisons with data are shown in Figures 4 and 5, respectively, for the 336 rod bundles run numbers 718 and 722. The run No. 718 had a power level of 1.258 MW and pressure of 5.5 MP_a (799 psia). The run No. 722 had a power level of 1.264 MW and a pressure of 2.7 MP_a (395 psia).

C. Heat Transfer Coefficient

The primary reason for tracking the mixture level is to determine heat transfer coefficients on the secondary side. The heat transfer coefficient for a U-tube node j is computed from:

$$h_j = \frac{(z_{top} - z_{mix})\, h_{nb} - (z_{mix} - z_{bottom})\, h_{dry}}{z\,(z_{top} - z_{bottom})} \quad \text{if } z_{bottom} < z_{mix} < z_{top}$$

$$= h_{nb} \quad \text{if } z_{mix} > z_{top}$$

$$= h_{dry} \quad \text{if } z_{mix} < z_{bottom} \tag{38}$$

where

z_{top} = elevation of the top of node j ,
z_{bottom} = elevation of the bottom of node j ,
z_{mix} = mixture elevation ,
h_{nb} = two phase boiling heat transfer coefficient,
h_{dry} = heat transfer cofficient after dryout.

The two phase heat transfer coefficent (h_{nb}) is computed from the Thom (6) correlation. The post-dryout heat transfer coefficient (h_{dry}) is input by the user.

VI. ANALYSIS OF A TYPICAL LOFW ATWS TRANSIENT

The LOFW ATWS transient analyzed was for a two loop PWR operating at 2610 MW (thermal). Pertinent plant parameters are summarized in Table I. The 2 group decay constants were obtained from the standard 6 group constants as follows:

$$\langle\lambda\rangle_1 = \left[\frac{1}{3}\sum_{i=1}^{3}\frac{\beta_i}{\lambda_i}\right]^{-1},$$

$$\langle\lambda\rangle_2 = \left[\frac{1}{3}\sum_{i=4}^{6}\frac{\beta_i}{\lambda_i}\right]^{-1}$$

Results of the base case calculation are presented in Figure 6. All feedwater is shut off at 0.0 seconds. The secondary system liquid inventory then begins to boil-off. At around 80 seconds the steam generator is completely dry and the primary system pressure and temperature begin rising sharply. The pressurizer relief valve opens at 75 seconds and the safety valve at 80 seconds. The pressurizer fills at 90 seconds and reaches a peak pressure of 33.7 Mpa (4896 psia) at 115 seconds.

Results of the base case analysis are compared with a RETRAN calculation (7), employing the same input parameters as in Table I, in Figures 7 through 9. The two sets of results agree fairly well, the major difference being a somewhat higher predicted peak pressure by the code developed here.

A number of cases were run to determine the sensitivity of the peak primary system pressure to various parameters. The results are summarized in Table II. The largest changes in peak pressure occurred due to variations in the moderator temperature coefficient, the Doppler coefficient, the fuel-clad gap conductivity, and the pressurizer relief and safety valve flows.

A. Moderator Temperature and Doppler Coefficents

Figure 10 illustrates variations in core power and primary system pressure due to changes in the moderator temperature coefficient (MTC). These plots illustrate the sensitivity of the peak pressure to the rate at which core power is reduced. Halving the MTC resulted in a peak pressure of 42.8 MP_a (6220 psia) at 122 seconds. Doubling the MTC reduced the peak pressure to 27.9 MPa (4050 psia).

Figure 11 illustrates the sensitivity of core power and primary system pressure to the Dopper coefficient. Note that doubling the coefficient slowed the rate of power decrease and hence resulted in a higher peak pressure. This is because the fuel temperature was dropping, thus adding positive reactivity.

B. Valve Flow Area

The effect of varying the valve flow areas on system pressure is illustrated in Figure 12. Assuming one of the two pressurizer relief valves fails to open increased the peak pressure to 41.3 MPa (6000 psia). Increasing the relief valve flow area by 50% (i.e. adding another relief valve) reduced the peak pressure to 29.2 MPa (4250 psia).

C. Gap Conductance

Table II indicates that the peak primary pressure is sensitive to the fuel-clad gap conductance. Halving the conductance increased the peak pressure by 7 MPa (1000 psia). This is due to the relationship between core power and fuel temperature. The lower gap conductance results in higher initial fuel temperature. As the core power drops, the decrease in fuel temperature is greater than that for the base case. This results in a greater positive reactivity addition through the Doppler feedback, hence a slower rate of power decrease and greater peak pressure.

VII. SUMMARY AND CONCLUSION

We have developed a relatively simple code for the analysis of anticipated transients without scram (ATWS) in a PWR. In particular, the loss of feedwater (LOFW) transient in which the secondary side of the steam generators boils dry in a short time has been treated. This has required tracking of the two phase mixture level in the secondary side and the attendant variation of the heat transfer rate from the primary to the secondary side. The modeling for the two phase mixture level is based on the drift flux model and was verified by comparing to the measured data in the separate effects boil-off experiments performed in the FLECHT SEASET and the 336 rod bundle facilities.

The basic approach employed in developing the code is lumped parameter modeling of the system components based on mass and energy conservation equations. The primary system during the LOFW ATWS transient remains full and does not experience phase separation and single phase description is adequate. The pressurizer is treated with two (liquid and vapor) non equilibrium regions with provisions for flow out of the relief and safety valves and the spray and heating for pressure control. The steam and subcooled water properties employed were polynomial fits shown in the RETRAN-2 manual. (1)

The code developed was employed in analyzing the LOFW ATWS transient for a typical PWR and the results obtained were compared with those obtained with the RETRAN-2 code employing the same input parameters. Good comparisons were obtained, thereby lending confidence to the basic soundness of the developed code. The main attractive feature, of course, is that the computer run time for the code developed here is about 1/15th of the transient real time. Thus, this code can be employed for extensive parametric analyses with little cost.

It is possible to extend the range of applications of the code developed here with selective improvements in modeling. For example, overcooling transients could be treated with minor modifications of the steam generator model. Similarly, it is possible to extend the primary system model to include tracking of a two phase mixture level in the core and thereby treat core undercooling transients. The main emphasis should be to employ as sophisticated (and complex) modeling as necessary to describe the transients adequately, with as little computational cost as possible, thereby providing a tool for rapid parametric analyses.

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Table 1
Input Parameters

Thermal Power	2610 MW
Moderator temperature coefficient	-9.9×10^{-5} ΔK/K/°C
Doppler coefficient	-1.9×10^{-5} ΔK/K/°C
Primary system volume	267 m^3 (9440 ft^3)
Pressurizer volume	42 m^3 (1500 ft^3)
Relief valve flow area	0.0020 m^2 (.022 ft^2)
Flow area multiplier	0.74
Safety valve flow area	0.0036 m^2 (0.039 ft^2)
Flow area multiplier	0.78
Initial secondary liquid inventory	119,000 Kg (262,000 lbm)
Void correlation	Drift flux
Kinetics parameters	
Prompt neutron lifetime (Λ)	0.00006 sec^{-1}
Group 1 delayed neutron fraction (β_1)	0.003647
Group 2 delayed neutron fraction (β_2)	0.002854
Group 1 decay constant ($\langle\lambda\rangle_1$)	0.339 sec^{-1}
Group 2 decay constant ($\langle\lambda\rangle_2$)	0.043 sec^{-1}

Table II
Sensitivity Study Summary

Run #	Parameter Varied	Peak Pressure
1	Base Case	33.7 MPa (4896 psia)
2	Moderator Temp. Coeff. Halved	42.8 MPa (6220 psia)
3	Moderator Temp. Coeff. Doubled	27.9 MPa (4053 psia)
4	Doppler Coeff. Halved	30.2 MPa (4382 psia)
5	Doppler Coeff. Doubled	40.8 MPa (5923 psia)
6	Gap Conductivity Halved	38.5 MPa (5582 psia)
7	Gap Conductivity Doubled	30.9 MPa (4488 psia)
8	Relief Valve Flow Areas Reduced by 50%	41.4 MPa (4488 psia)
9	Relief Valve Flow Areas Increased by 50%	29.3 MPa (4256 psia)
10	No Spray Flow	33.8 MPa (4901 psia)
11	No Aux Feed	36.5 MPA (5293 psia)
12	Turbine Trip at 10 sec	330. MPa (4793 psia)

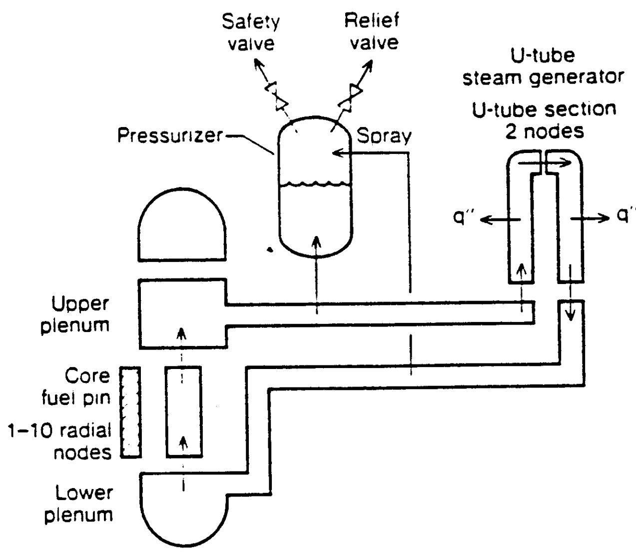

Figure 1. Primary system model.

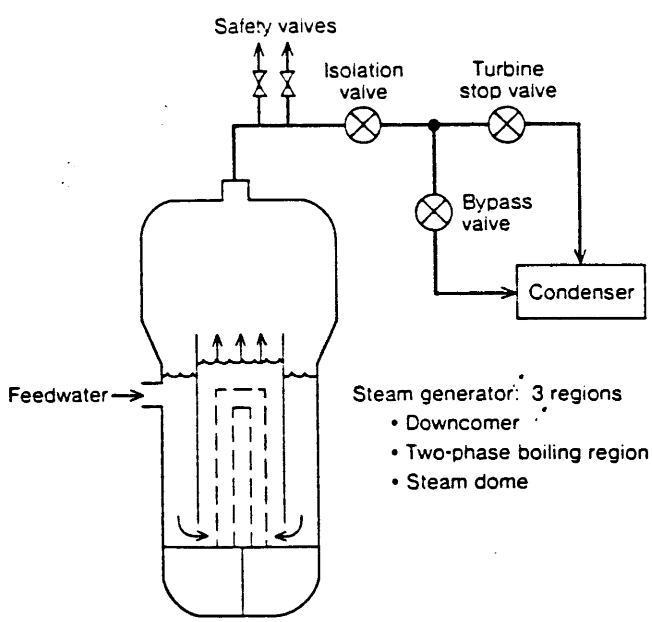

Figure 2. Secondary system model.

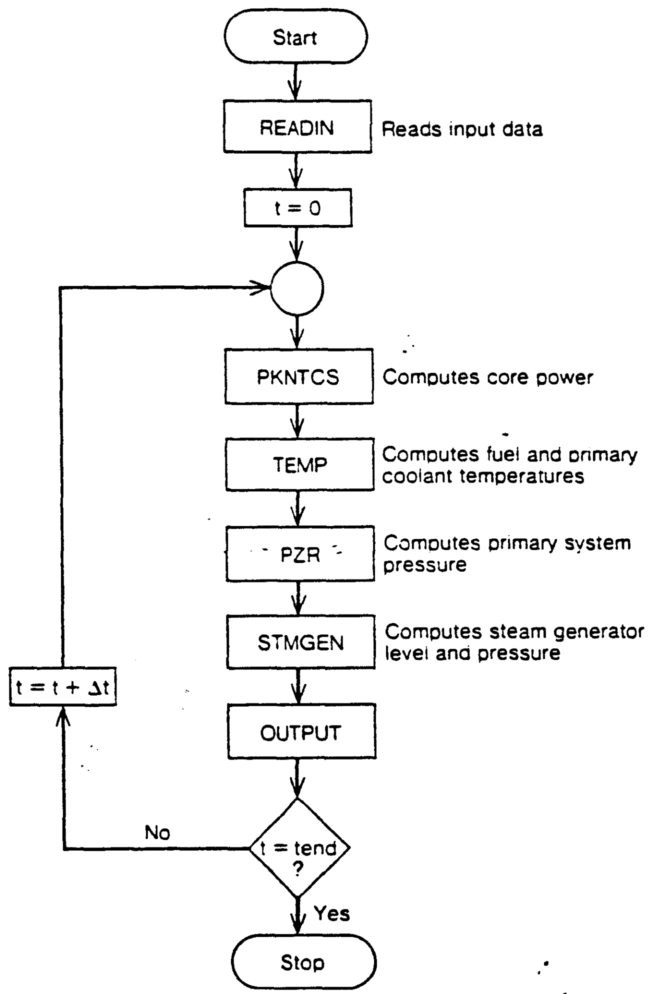

Figure 3. Flow chart of main program.

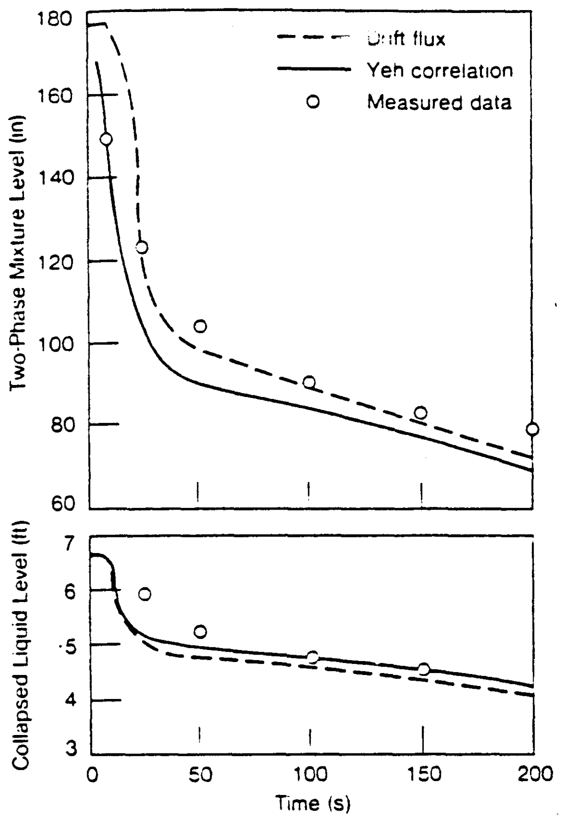

Figure 4. 336 rod bundle, run number 718.

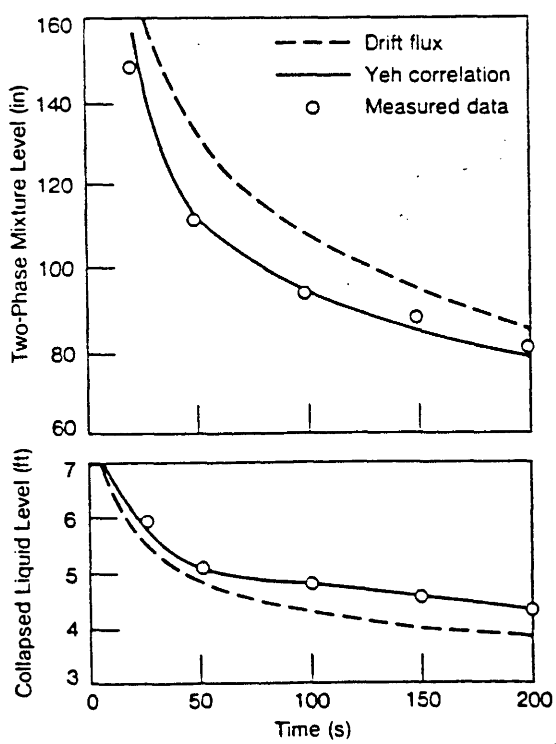

Figure 5. 336 Rod bundle run 722.

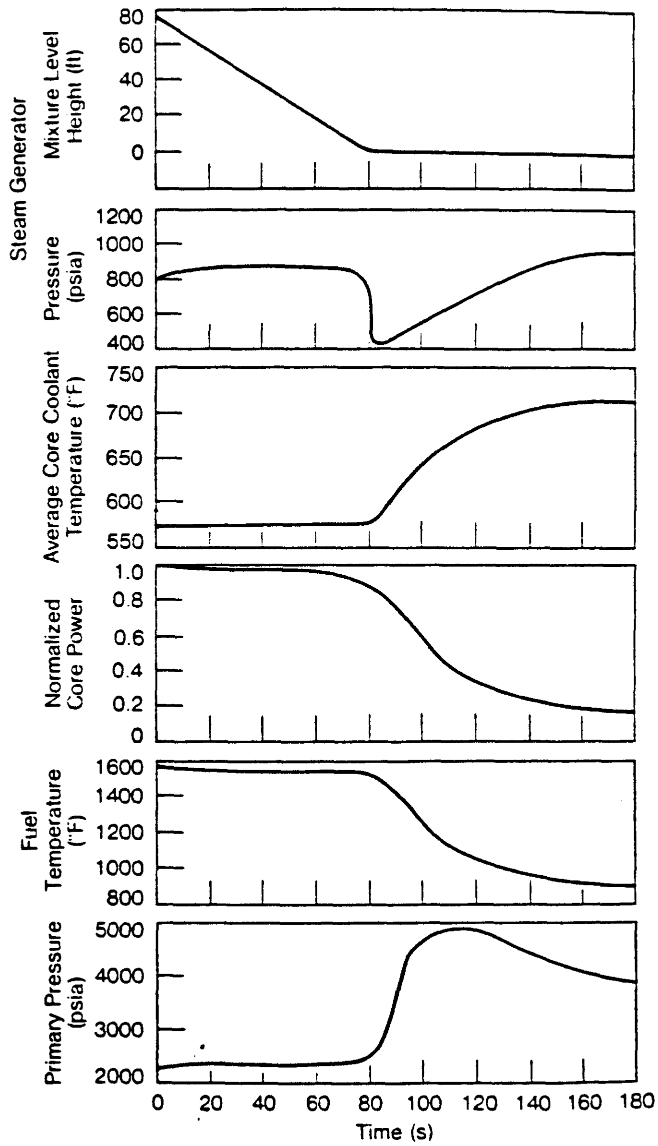

Figure 6. LOFW ATWS TRANSIENT results of base case.

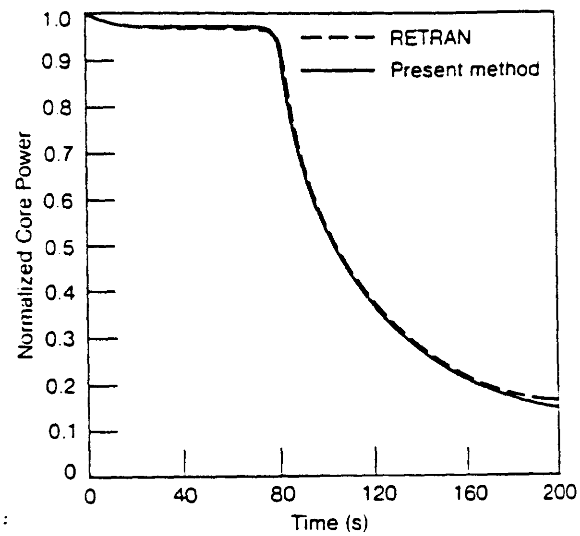

Figure 7. Comparison of normalized core power.

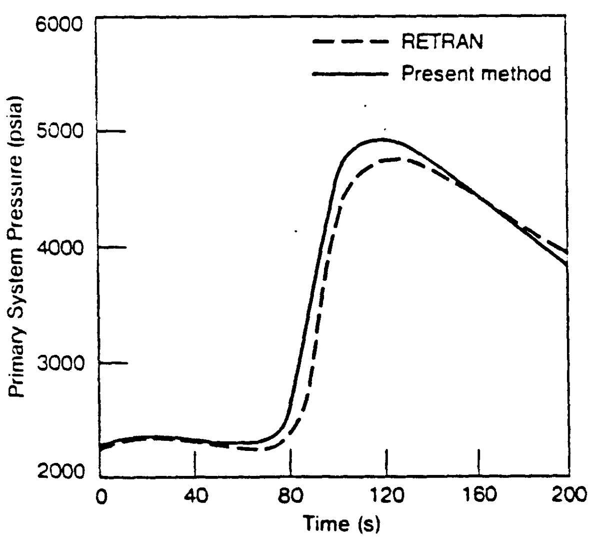

Figure 8. Comparison of primary system pressure.

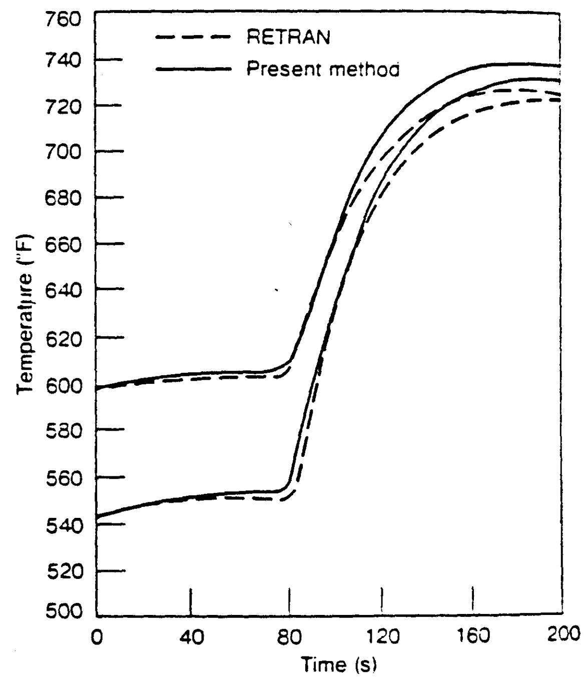

Figure 9. Comparison of hot and cold leg temperatures.

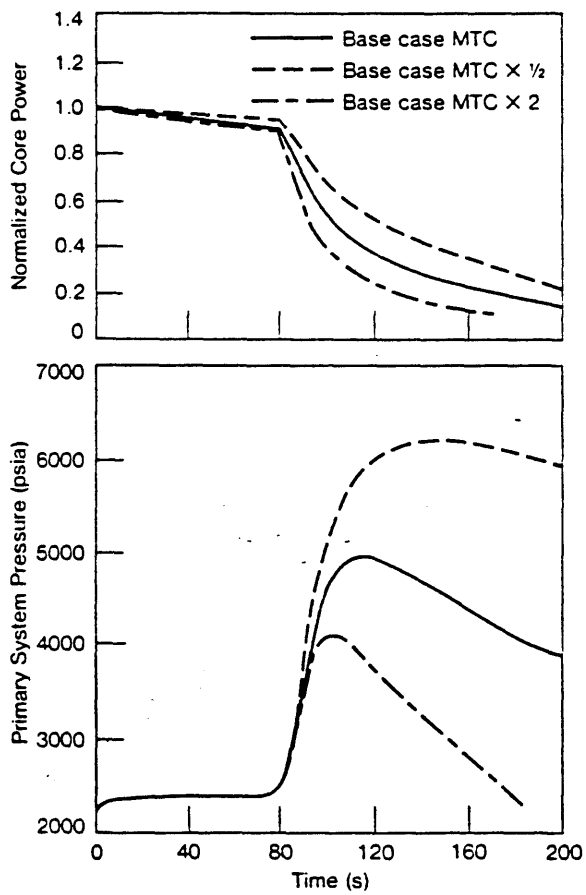

Figure 10. Sensitivity of primary system pressure to moderator temperature coefficient (MTC).

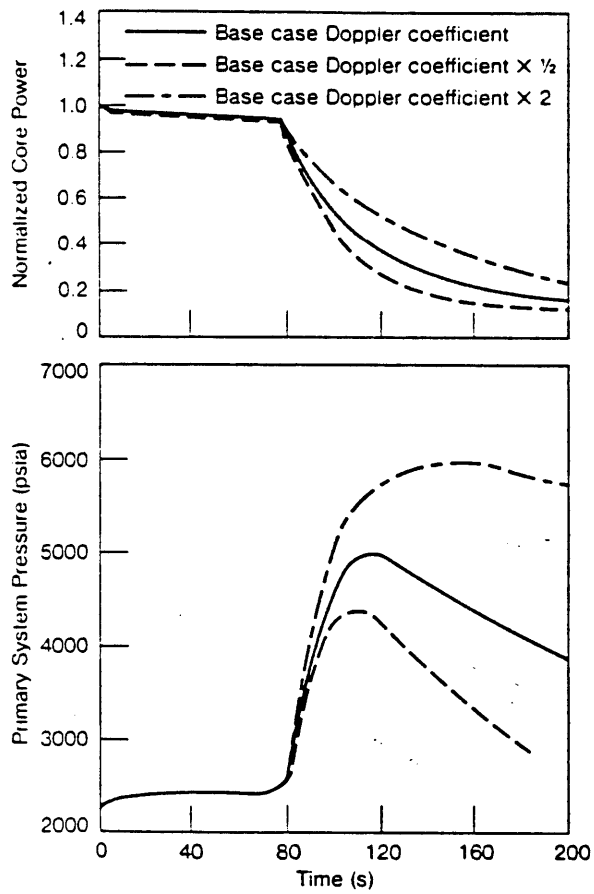

Figure 11. Sensitivity of primary system pressure to Doppler coefficient.

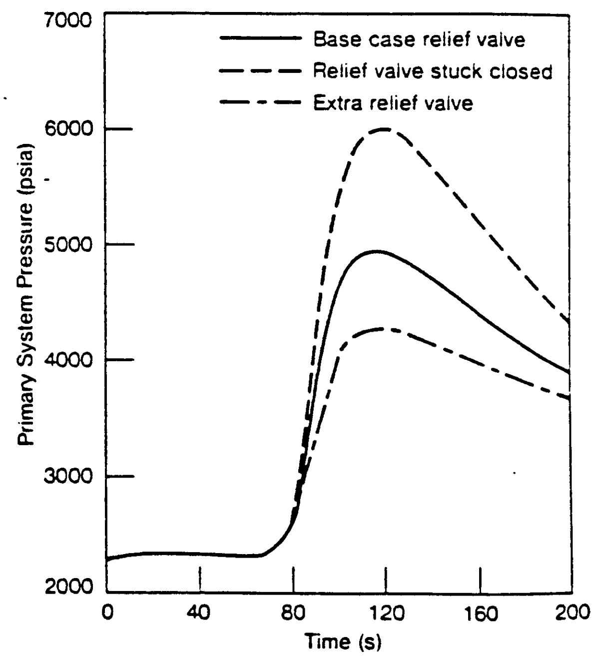

Figure 12. Sensitivity of primary system pressure to relief valve flow area.

ANNEX B

FATIGUE CRACK GROWTH OF ALLOYS X-750 AND 600 IN SIMULATED PWR AND BWR ENVIRONMENTS

By

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R.M. Latanision[(3)], and R.M.N. Pelloux[(3)]

Extended Abstract
Submitted to

International Symposium on Environmental
Degradation of Materials
in Nuclear Power Systems - Water Reactors

August 22-24, 1983
Myrtle Beach Hilton Hotel
Myrtle Beach, South Carolina

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(3) Professor, Mass. Institute of Technology, Cambridge, MA, USA

Nickel-base alloys are used extensively in Light Water Reactor (LWR) nuclear power systems due to their excellent corrosion resistance and strength in high temperature aqueous environments. Alloy 600, a solid solution strengthened nickel-base alloy, is used for Pressurized Water Reactor (PWR) steam generator tubing for almost all U.S. PWR power stations. Alloy 600 is also used in Boiling Water Reactor (BWR) systems for safe ends and other components. For applications requiring very high strength and superior corrosion resistance, age hardenable nickel-base alloys such as Alloy X-750 and Alloy 718 are often used. Specific applications include fuel assembly hold down springs, BWR Jet Pump components and high strength bolts for core and other structural components. In spite of their generally excellent general corrosion resistance the above mentioned alloys have been found to be susceptible to localized forms of corrosion attack such as stress corrosion cracking, corrosion fatigue, and intergranular attack. Susceptibility to these forms of damage is a function of prior thermomechanical treatment and service environment. For Alloy 600, from an economic standpoint, the most severe impact has resulted from instances of intergranular attack (IGA) and denting related cracking initiated from the secondary side of PWR steam generators; there have been instances where cracking from the primary side environment has occured. In some of these cases, fatigue could not be ruled out as a contributing factor. For certain environments and heat treatments Alloy X-750 is susceptible to environmentally assisted fatigue and stress corrosion cracking. Failures of components in service, while few in number, have resulted in significant economic impact. These failures have provided impetus for the development of materials with improved resistance to these environmental effects but the available data is limited. For this reason a research program, under the sponsorship of the Electric Power Research

Institute, has been initiated with the goal of achieving a mechanistic understanding of the fatigue and stress corrosion cracking behavior of Ni-Cr-Fe alloys used in nuclear power systems. This paper presents results of a program to evaluate the effect of thermomechanical processing, environmental and mechanical variables on the corrosion fatigue behavior of Alloys X-750 and 600.

Alloy X-750

The Alloy X-750 used in the program was tested in three metallurgical conditions. These conditions were selected based on a desire to evaluate material conditions prototypic of that used for components of past, present and future vintage. Material heat treatments were as follows:

- Equalized and Aged (AH)
  Hot Worked + 24 Hr @ 885^{o}C/AC + 20 Hr @ 704^{o}C/AC
- Low Temperature Anneal (BH)
  Hot Worked + 1 Hr @ 982^{o}C/AC + 20 Hr @ 704^{o}C/AC
- High Temperature Anneal (HTH)
  Hot Worked + 1 Hr @ 1093^{o}C/AC + 20 Hr @ 704^{o}C/AC

The above heat treatments result in significant microstructural differences. The microstructure of the AH material is characterized by a bi-modal γ' size distribution with a grain size of approximately 15 μm. The grain boundaries are decorated with carbides as well as containing γ' of the larger size fraction in the bi-modal distribution. The intragranular region is characterized by both a large and small γ' size fraction. Regions adjacent to the grain boundaries are free of the large γ' size fraction. The BH material is characterized by a bi-modal grain size distribution and a single size fraction of γ'. The grain size for the larger size fraction is approximately 150 μm, while that for the smaller size fraction is approximately 150 μm, while that for the smaller size fraction is approximately 15 μm, similar to that for the AH material. The HTH material is characterized by a large grain size, 150 μm, and a single line γ' size fraction.

Fatigue tests were conducted at frequencies of 0.1 and 10 Hz using a sine wave loading history with a stress rate, P_{min}/P_{max}, of 0.1 in the following environments: (1) Air @ 25^{o}C, (2) High Purity Oxygenated Water (8 ppm) @ 93,288,316^{o}C, (3) High Purity Deoxygenated Water (<5 ppb), (4) Simulated BWR water chemistry with hydrogen additions, and (5) Simulated PWR primary water chemistry. Testing was conducted using CT specimens with 1T dimensions except for specimen thickness, which was 12.7 mm. Tests were performed in a titanium autoclave system integrated with a fully automated servohydraulic fatigue machine. Specimens were electrically isolated from the load train grips and autoclave. Electrochemical potential was monitored during testing using an external Silver/Silver Chloride reference electrode. Crack length was evaluated using compliance measured using an in-situ LVDT. Crack growth rate data was collected using crack length increments at constant ΔK.

Figures 1-3 present some of the experimental results. Crack growth rates are a function of ΔK, frequency, heat treatment, and environment. In addition, crack morphology is also a function of the above variables. Frequency effects are the most significant for the AH heat treatment and decrease in significance for the BH treatment and still further for the HTH heat treatment. With minor exceptions, crack growth rates decrease, for a given ΔK, from AH to BH to HTH.

Figure 1 shows some experimental results for the AH material. Several points should be noted: (1) the crack growth rate is a strong function of temperature and frequency, (2) the crack path is a function of ΔK, temperature and frequency. At $93^{o}C$ the crack path parallels the grain boundaries at 10 Hz but switches to a more transgranular mode at 0.1 Hz. At 288 and $316^{o}C$ the crack path is mostly transgranular at both frequencies. For values of ΔK greater than 25 $MPa\sqrt{m}$ the crack path becomes more transgranular and striated. The effect of frequency reverses itself between 93 and $288/316^{o}C$.

Figure 2 shows data obtained to date for the BH heat treatment. Crack growth rates are lower than those for the AH treatment. At low values of ΔK the crack path is intergranular within the small grain size fraction and transgranular but strongly crystallographic within the large grain size fraction at all temperatures. At higher ΔK's the fracture path becomes more transgranular and striated for both size fractions. The frequency effects, observed in the AH treatment are still present but to a lesser extent.

Figure 3 shows data obtained to date for the HTH treatment. Crack growth rates are generally lower than the other heat treatments and, while the frequency effects are still present, they are minimal. Crack paths are strongly crystallographic and transgranular at lower ΔKs but more striated and less crystallographic at high ΔKs.

Alloy 600

Material for the Alloy 600 test program consisted of plate material with a thickness of 12.7 mm. The processing schedule for the plate was designed, in consort with the vendor, to produce a microstructure similar, both in terms of mechanical and corrosion behavior, to that for product line mill-annealed steam generator tubing of the type used in older

Westinghouse NSS systems. Analysis indicated that this goal was achieved. Grain size, carbide morphology and corrosion behavior are consistent with tubing. Tests were conducted at room temperature and 288°C at frequencies of from 0.1-10 Hz. Two basic heat treatments were employed in the program: (1) mill annealed plus a 2 hour age @ 700°C and (2) mill annealed plus a 120 hour age @ 700°C. The two hour age produced a highly "sensitized" condition as evidenced by severe chromium depletion at the grain boundaries. Material thermally treated for 120 hours @ 700°C exhibited no chromium depletion at grain boundaries. Sensitization behavior of the Alloy 600 material was examined and quantified using conventional corrosion testing and high resolution scanning transmission electron microscopy (STEM) using a Vacuum Generators, Inc., HB-5 STEM. Detailed analysis of grain boundary and near grain boundary analysis were performed.

Test environments used in the test program included the following: (1) room temperature air @ 25°C, (2) 288°C, high purity water, 8 ppm oxygen, (3) 288°C simulated secondary AVT chemistry and (4) 288°C high purity low oxygen water.

Figures 1 and 2 present the data generated in the program thus far. The overall test matrix is incomplete at this time; but, so far, the data indicates that there is little, if any, effect of thermal treatment on the crack growth rate. There is, however, a significant frequency effect as well as a stress ratio, (P_{min}/P_{max}), effect. The crack path, in both cases, is transgranular except at low (<10 mPz$\sqrt{m}$) ΔKs. At low ΔKs the crack path becomes increasingly intergranular.

Figure 1

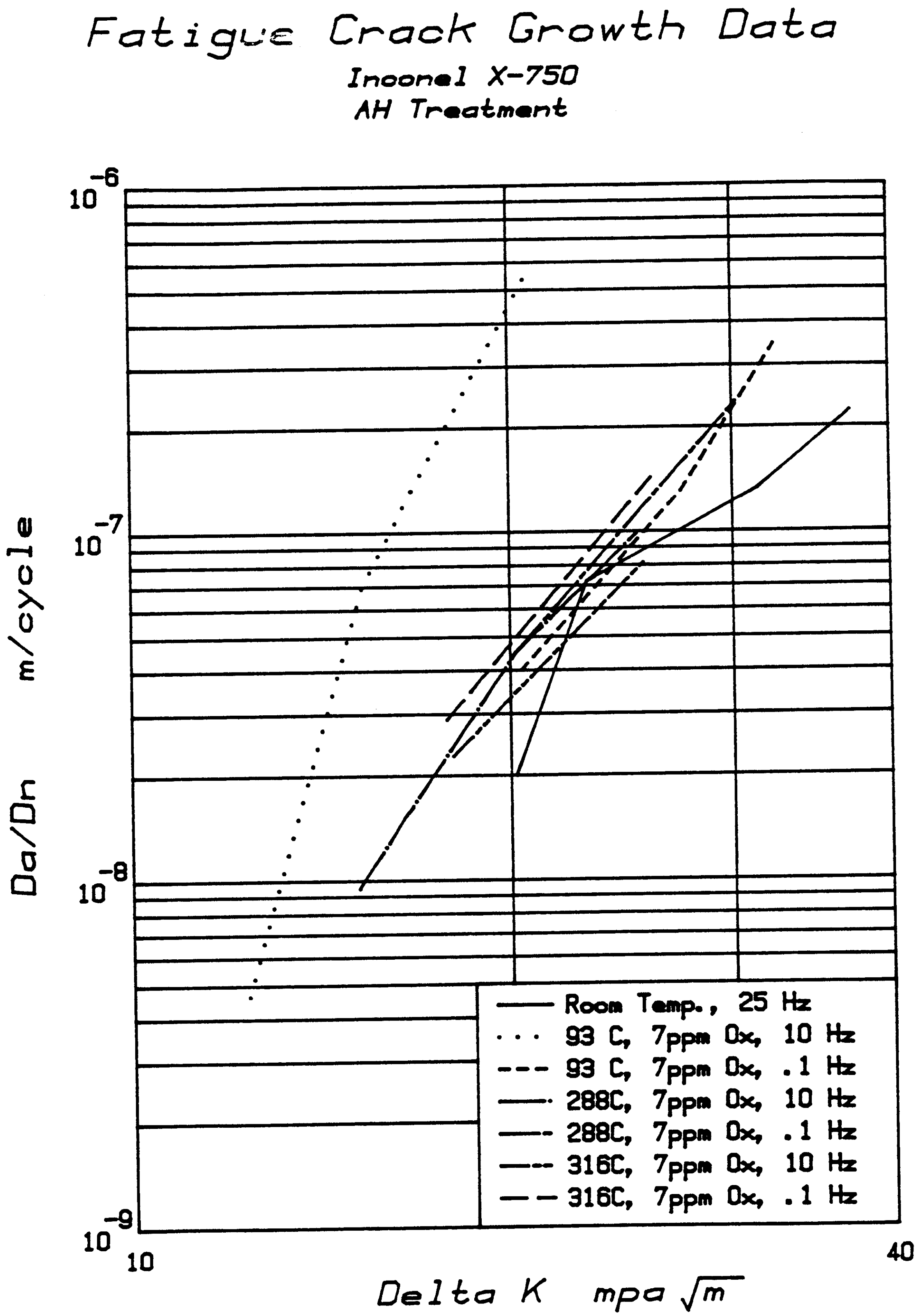

Fatigue Crack Growth Data
Inoonel X-750
AH Treatment
10^-6
10^-7
10^-8
10^-9
Da/Dn m/cycle
10
40
Delta K mpa √m
Room Temp., 25 Hz
93 C, 7ppm Ox, 10 Hz
93 C, 7ppm Ox, .1 Hz
288C, 7ppm Ox, 10 Hz
288C, 7ppm Ox, .1 Hz
316C, 7ppm Ox, 10 Hz
316C, 7ppm Ox, .1 Hz

Figure 2

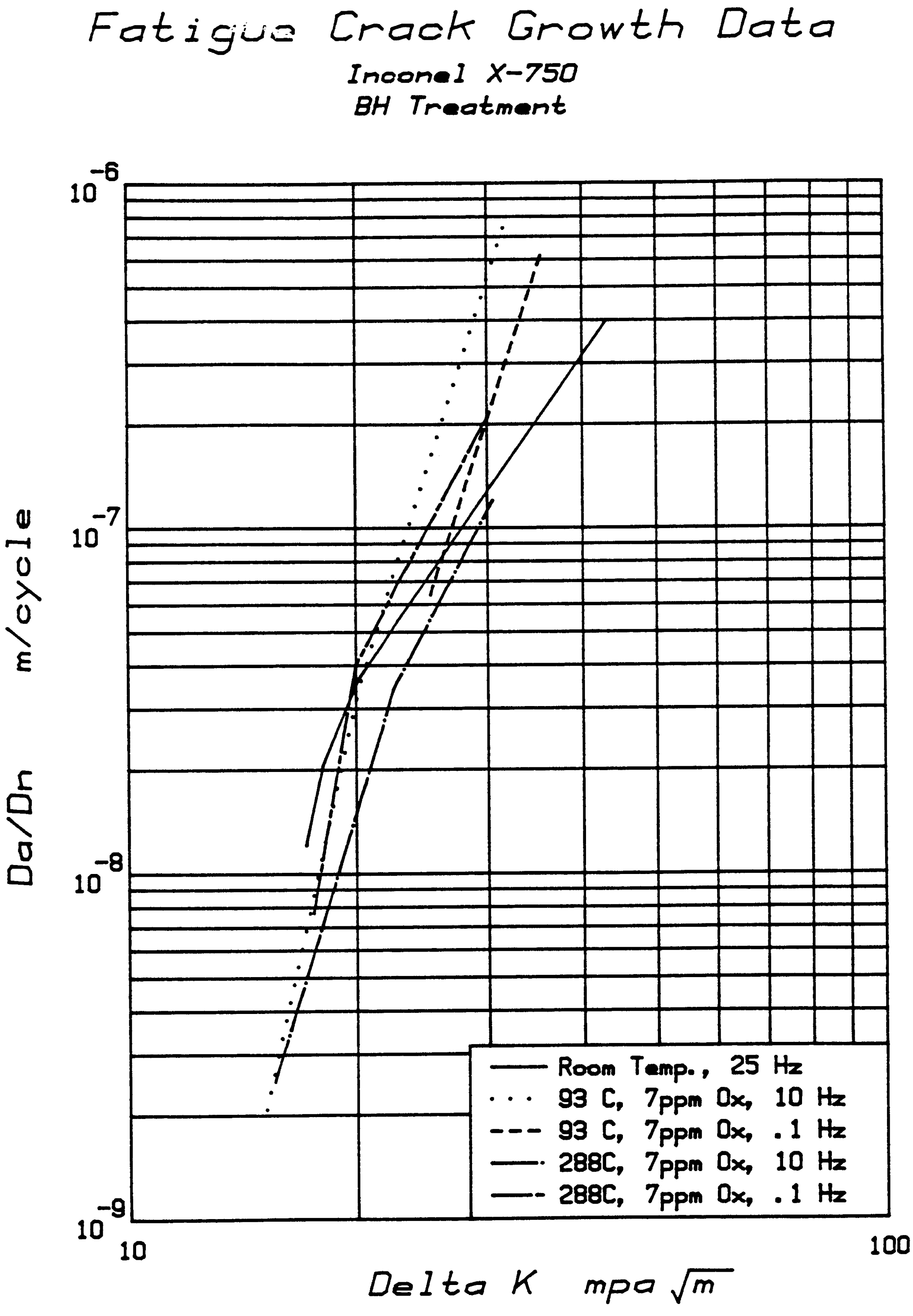

Fatigue Crack Growth Data
Inconel X-750
BH Treatment
10^-6
10^-7
10^-8
10^-9
10
100
Da/Dn m/cycle
Delta K mpa √m
Room Temp., 25 Hz
93 C, 7ppm Ox, 10 Hz
93 C, 7ppm Ox, .1 Hz
288C, 7ppm Ox, 10 Hz
288C, 7ppm Ox, .1 Hz

Figure 3

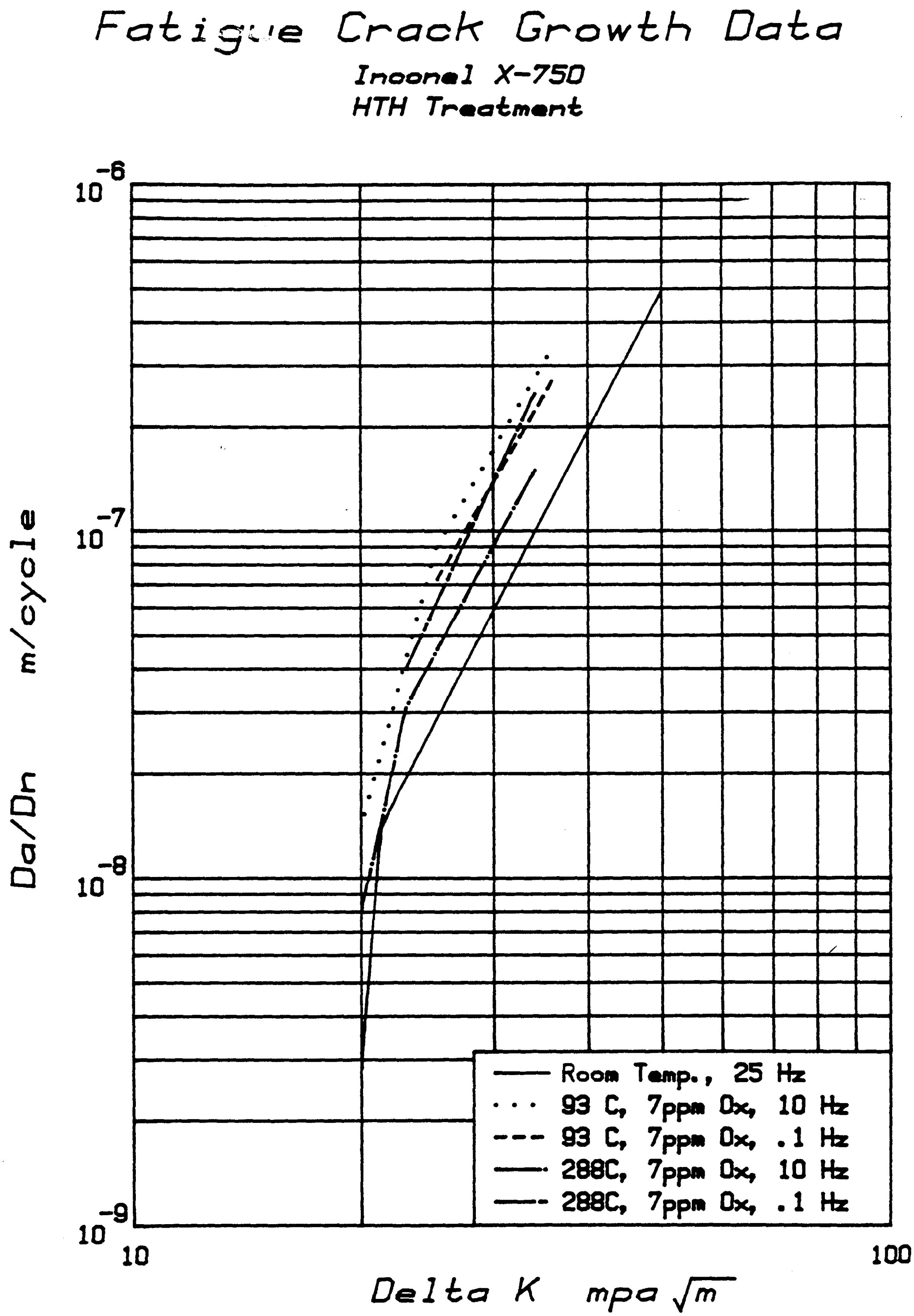

Fatigue Crack Growth Data
Inconel X-750
HTH Treatment
10^-6
10^-7
10^-8
10^-9
10
100
Da/Dn m/cycle
Delta K mpa √m
Room Temp., 25 Hz
93 C, 7ppm Ox, 10 Hz
93 C, 7ppm Ox, .1 Hz
288C, 7ppm Ox, 10 Hz
288C, 7ppm Ox, .1 Hz

Figure 4

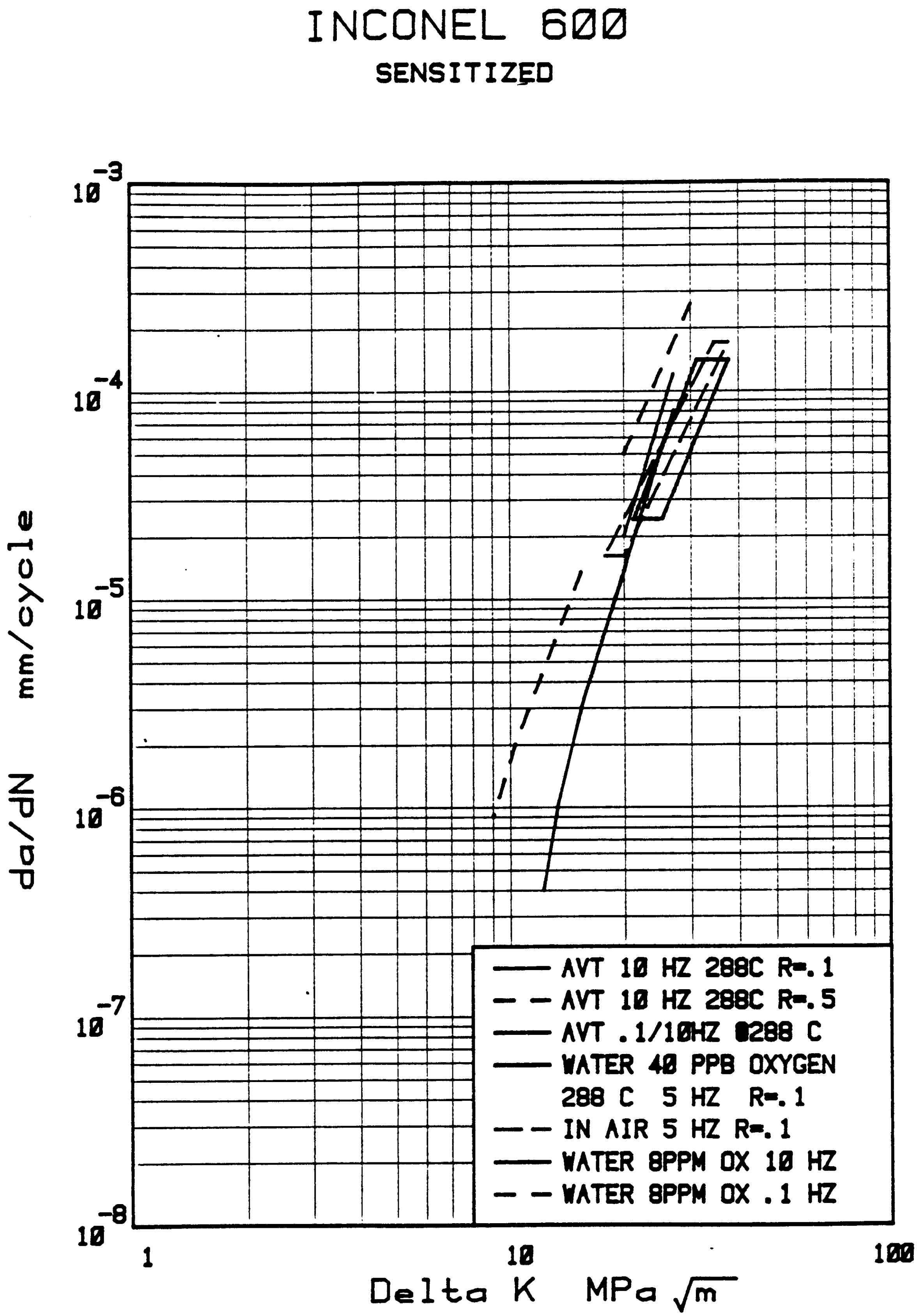
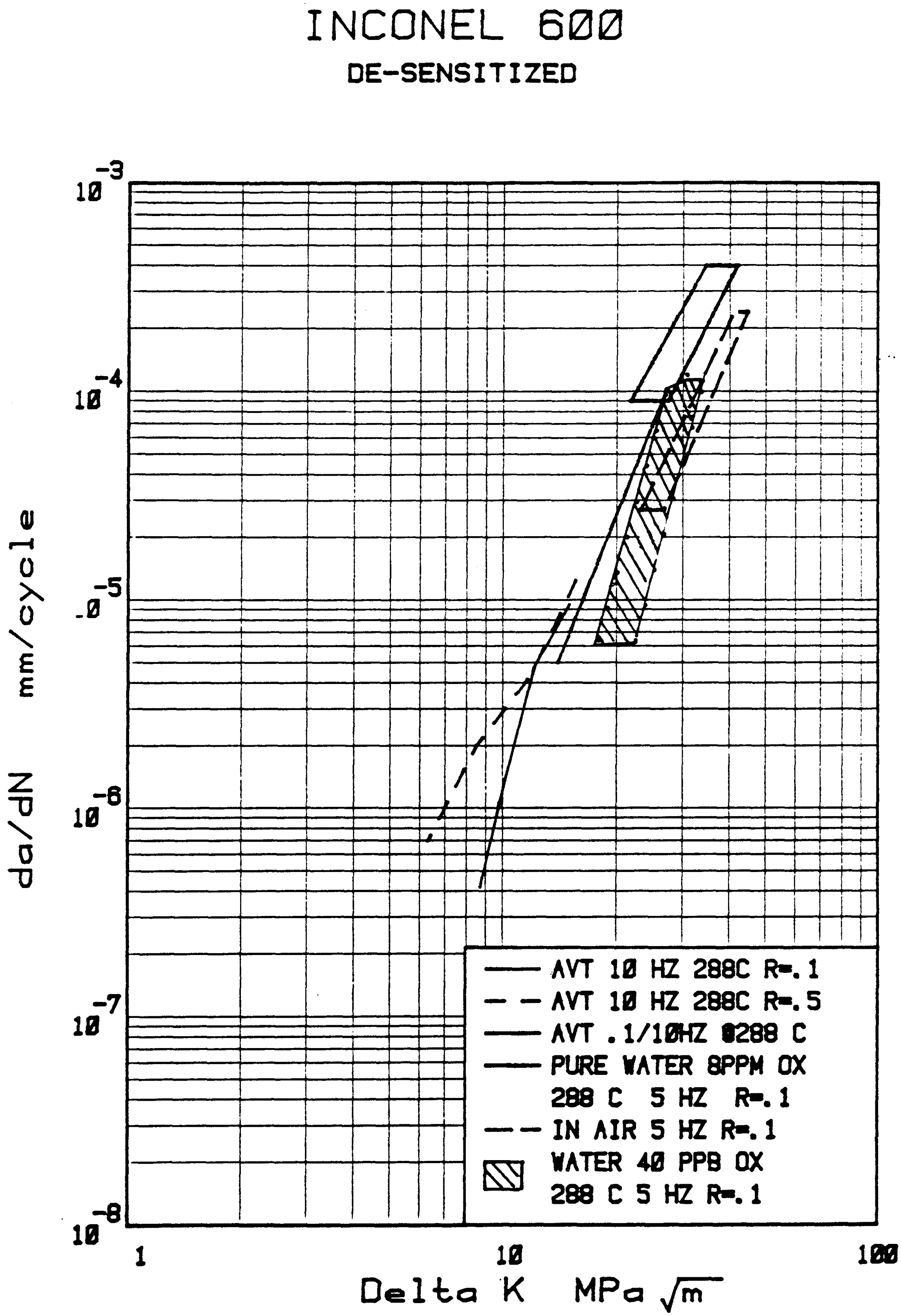

INCONEL 600
SENSITIZED
10^-3
10^-4
10^-5
10^-6
10^-7
10^-8
da/dN mm/cycle
1
10
100
Delta K MPa √m
AVT 10 HZ 288C R=.1
AVT 10 HZ 288C R=.5
AVT .1/10HZ @288 C
WATER 40 PPB OXYGEN
288 C 5 HZ R=.1
IN AIR 5 HZ R=.1
WATER 8PPM OX 10 HZ
WATER 8PPM OX .1 HZ

Figure 5


INCONEL 600
DE-SENSITIZED
da/dN mm/cycle
10^-3
10^-4
10^-5
10^-6
10^-7
10^-8
1
10
100
Delta K MPa √m
AVT 10 HZ 288C R=.1
AVT 10 HZ 288C R=.5
AVT .1/10HZ @288 C
PURE WATER 8PPM OX
288 C 5 HZ R=.1
IN AIR 5 HZ R=.1
WATER 40 PPB OX
288 C 5 HZ R=.1

ANNEX C

American Control Conference, San Fransisco, CA, June 1983

DIGITAL CONTROLLER FOR A NUCLEAR REACTOR

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Abstract

This paper presents the development and implementation of a digital control system in an operating nuclear reactor. The control system incorporates on-line detection and isolation of faulty equipment, sensor calibration, measurement estimation, and information display in a given controller structure. It is tolerant of process disturbances, certain equipment failures, and sensor degradation and noise.

Rather than relying on signals from single sensors, each feedback/feedforward signal that is sent to the controller under this approach is a digitally processed, weighted average of several valid measurements of the appropriate process variable. The weights are not à priori fixed but depend on the à posteriori probabilities of failure of individual sensors and are computed on the basis of past and current observations. Thus, for a gradually degrading measurement, its weight is smoothly reduced and eventual isolation of the fault does not cause an abrupt change in the estimate, i.e. the controller action remains bumpless.

Introduction

In contrast to avionic and industrial processes, the role of computers in the instrumentation and control of nuclear power plants, excepting the Canadian HWR units, has been quite limited [1]. Recently, computer-aided fault diagnosis and information display techniques have been developed to enhance the safety of nuclear power plants [2-6]. It is possible to achieve safety, reliability, operational flexibility, and superior performance by simultaneously applying on-line fault detection and isolation (FDI), sensor calibration, and information display techniques to a feedback/feedforward control system. An integrated control system that has inherent capabilities to perform the afore-mentioned tasks has been developed and designed to be executable by commercially available microcomputers. This control system has been implemented in an on-going series of experiments on the 5 MWt nuclear research reactor, MITR-II, which is licensed by the Nuclear Regulatory Commission (NRC) and is operated by Massachusetts Institute of Technology. Presently, the power of the MITR-II is being digitally controlled, under both steady-state and transient operations, via feedback of an on-line estimate from a set of power sensors. A real-time CRT display presents the validated data and diagnostics of the power-related instrumentation such as neutron flux detectors, primary coolant flow, and temperature sensors.

The major functions of this on-line control system relative to the MIT nuclear research reactor are:

(1) The regulation of reactor power under both steady-state and transient conditions. The latter include xenon oscillations, coolant temperature variations, and operator-induced load changes.

(2) The on-line detection, isolation, and reconfiguration of faulty sensors without interruption of plant operation.

(3) The on-line estimation of both measurable and non-measurable plant variables such as power, coolant flow, temperature, and reactivity using the available sensors and/or analytic redundancies [7] that rely on the physical relationships amongst the plant variables.

(4) The on-line calibration of power sensors to compensate for process disturbances such as changes in the spatial distribution of the neutron flux due to xenon transients and the resulting shim blade motion.

(5) The on-line information display of both the critical plant variables and the diagnostics of any faulty sensors and equipment, thereby assisting the operators to make timely and appropriate decisions.

This paper is organized in several sections that describe the design concepts, the experimental facility, the pertinent results, and the experimentally-deduced conclusions. While the digital control system has been demonstrated on an operational nuclear reactor, it is adaptable to large and complex continuous processes such as chemical and power plants.

Background of the Signal Validation Methodology

The signal validation methodology uses redundant measurements that may be obtained either directly, as sensor outputs, or indirectly as analytic calculations from a mathematical model formulated on the basis of physical relationships among other process variables (e.g. mass and energy balances in thermo-fluid processes). The methodology provides a unified, systematic procedure for (1) fault detection and isolation (FDI), and (2) sensor calibration and measurement estimation.

The FDI technique, used in the control system, is a sequential decision-making procedure that systematically seeks out the largest consistent subset from a set of redundant measurements where the consistencies among individual measurements of a given process variable are determined on the basis of allowable errors [2,3]. Allowable errors or error bounds, specific to individual measurements, can be obtained from either experimental data or the information on instrument tolerances. As the number of redundant measurements increases, the checking of consistency in all possible combinations and the attendant task of bookkeeping for all information at the current and past sampling instants becomes very complex. A systematic procedure, appropriate for a digital processor, that relies on recursive relations, based on the consistencies of each measurement relative to the remaining ones, has been developed for diagnosing sensor and plant equipment failures. The mathematical background and other details of the FDI technique are given in the previous publications [2,3].

The sensor calibration and measurement estimation technique is also a sequential procedure that is performed on-line in the framework of the aforesaid FDI technique. If the process variables being measured are time-dependent and if the redundant sensors are installed in different spatial locations (e.g., neutron flux detectors) or if analytic measurements [7] are used to supplement the sensor redundancy (e.g., fluid temperature obtained from thermal power balances), the individual measurements of the given process variable may exhibit deviations from each other after a length of time even though the sensors are functioning normally. These differences could be caused by time-varying plant parameters, reaction kinetics, transport delays, etc. Consequently, some of the sensors may be erroneously deleted unless they are periodically recalibrated. On the other hand, failure to isolate a degraded sensor could cause an inaccurate estimate of the measured variable. More importantly, the plant's performance might be adversely affected if that inaccurate estimate were used as an input to the controller. These difficulties can be circumvented if (1) only the consistent measurements are calibrated such that their residuals are minimized, and (2) the weights of the calibrated measurements (for computing the estimate) are updated on the basis of their respective à posteriori probabilities of failure instead of being à priori fixed. The technique is therefore capable of detecting both abrupt and slow failures. Specifically, in the event of abrupt disruptions in some sensor(s) in excess of the specified error bound(s), the respective sensor(s) are isolated by the FDI algorithm, and only the remaining sensors are calibrated and used to provide an estimate. If a gradual sensor degradation occurs, the faulty sensor may not be immediately isolated but its influence on the estimate and the calibration of the remaining sensors is diminished as a function of its degradation because its weight decreases with an increase in the à posteriori probability of failure. Thus, if the error bounds of the measurements are appropriately increased to reduce the probability of false alarms, the resulting delay in detecting a gradually degrading sensor could be tolerated because an undetected fault, as a result of its reduced weight, does not significantly affect the accuracy of calibration and estimation. Moreover, since the weight of a gradually degrading measurement for computing the estimate is smoothly reduced, the eventual isolation of the fault does not cause any abrupt change in the estimate, thereby assuring a bumpless controller action. Details of the calibration and estimation technique, and relevant experimental results are presented in another paper [8] at this conference.

Description of the Reactor and the Instrumentation

A description of the system configuration and instrumentation of the 5 MWt fission reactor is given in the MITR-II Reactor Systems Manual [9]. The reactor is heavy-water reflected, light-water moderated and cooled, and functions as a research and educational facility. A simplified diagram of the process and instrumentation is given in Figure 1. The nuclear instrumentation used for the research reported in this paper consists of three neutron flux sensors and a gamma-ray sensor that correlates neutron power with the radioactivity (N-16) of the primary coolant. Four independent measurements of primary coolant flow are obtained from the pressure differences across orifices. Primary coolant temperatures are measured as follows: two sensors for the hot leg, two sensors for the cold leg, and one sensor for temperature difference between the legs. In effect, three measurements are available for the temperature difference. The noise and statistical characteristics of the MITR-II's flow, temperature, and neutron flux instrumentation are similar to those in commercial reactors. These sensors are hardwired to a portable LSI-11/23 microcomputer through appropriate isolators, signal conditioners, and A/D converters. None of the sensors that form the MITR-II's safety system were used for this research.

At present, neither the computer nor the data acquisition and reconstruction system hardware is fault-tolerant. They consist of a single string processor, memory, real-time clock, and both A/D and D/A convert-

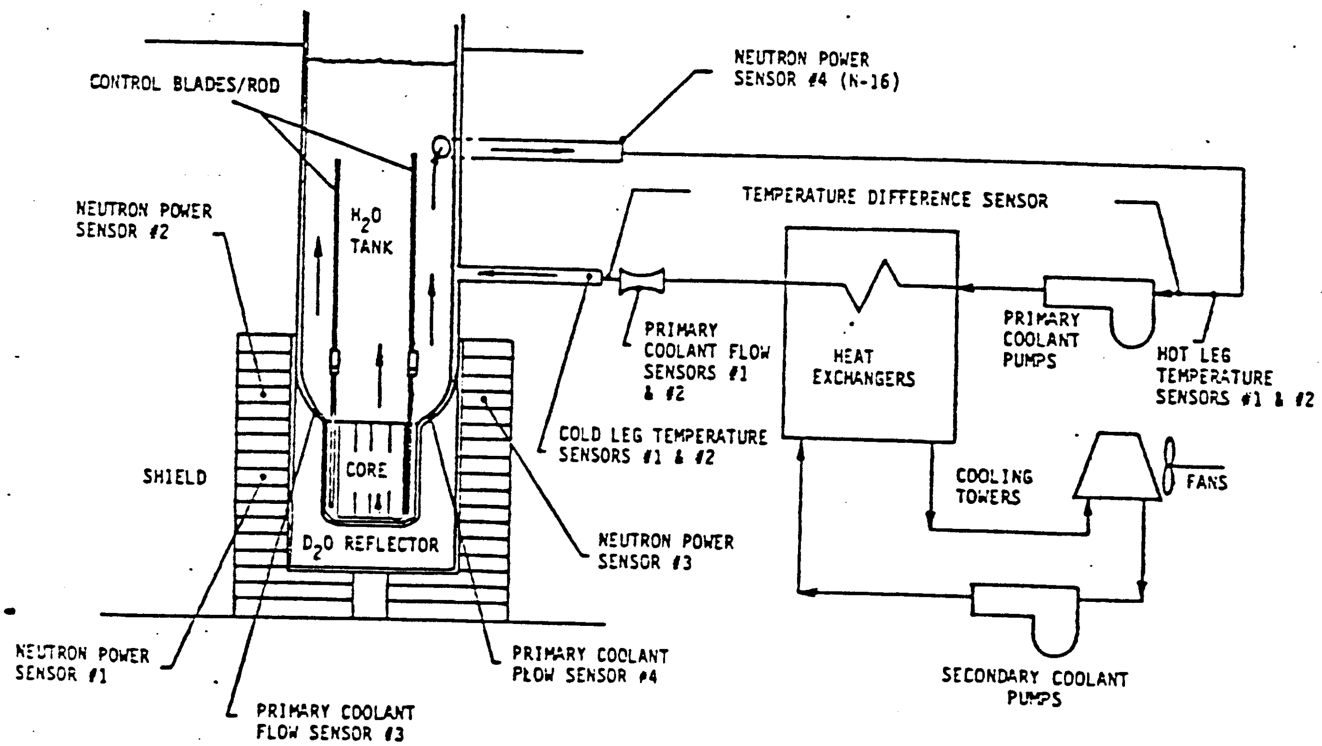

Figure 1. Simplified Process and Instrumentation Diagram for MITR-II

ers. The software is designed to be fault-free with a very high probability. It is structured as a main program with several modular subprograms following a top-down hierarchy such that an appropriate software test procedure can systematically identify the faulty modules. The software is also capable of detecting the majority of the hardware failures that are likely to occur in the course of operations. Details of the software architecture are not presented here due to space limitations.

Description of the Control Scheme

Coarse control of the power in the MITR-II is achieved by manually positioning a bank of six shim blades. Once critical, the neutron flux is normally maintained constant by an automatic analog controller that is monitored by the reactor operator. The controller adjusts a fine-control regulating rod according to the feedback signal of a single power sensor, as shown in Figure 2. If the error signal, i.e. the difference between the reference and the sensor signal exceeds a specified bound (typically around 2% of the rated reactor power of 5 MWt), "automatic" control is tripped to the "manual" mode in which control is maintained directly by the reactor operator. The analog controller in Figure 2 is a standard proportional-integral-derivative (P-I-D) type. The controller output energizes a 3-position relay to drive a constant speed motor that moves the regulating rod up or down at 108 mm/minute, which corresponds to an average of reactivity change of about $1.1 \times 10^{-5}\Delta K/K$ per second.

The digital control system shown in Figure 3, replaces the single power sensor in Figure 2 by the four available sensors, supplies the analog controller and its relay logic by a digital algorithm, and incorporates on-line FDI, sensor calibration, and information display. The constant-speed motor has also been replaced by a variable-speed stepping motor in order to implement continuous-action control laws. (Note: The maximum speed of the motor is limited to reduce the consequences of any unforeseen malfunction. Also, the relay logic of the analog controller can be replicated.) The structure of this digital control system is shown in Figure 4.

The multiply redundant sensors for power, flow, and temperature difference measurements are first validated on-line with the aid of the FDI methodology [2, 3]. The MITR-II's flow and temperature sensors are stable and do not require on-line calibrations. However, certain of the MITR-II's neutron power sensors may be affected by process disturbances. This occurs

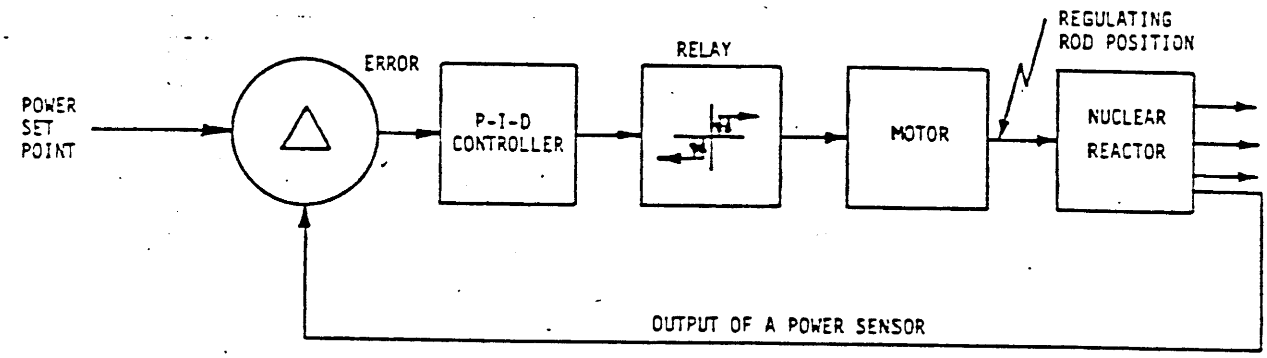

Figure 2. Analog Control Scheme

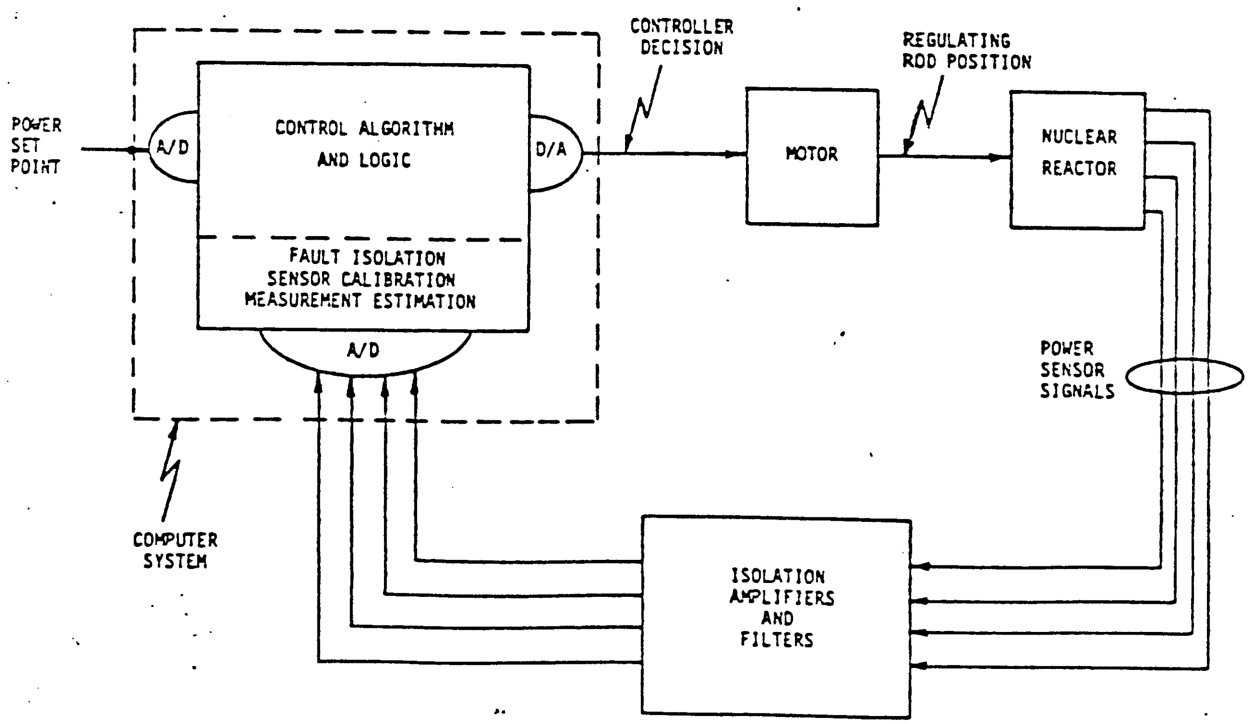

Figure 3. Digital Control Scheme

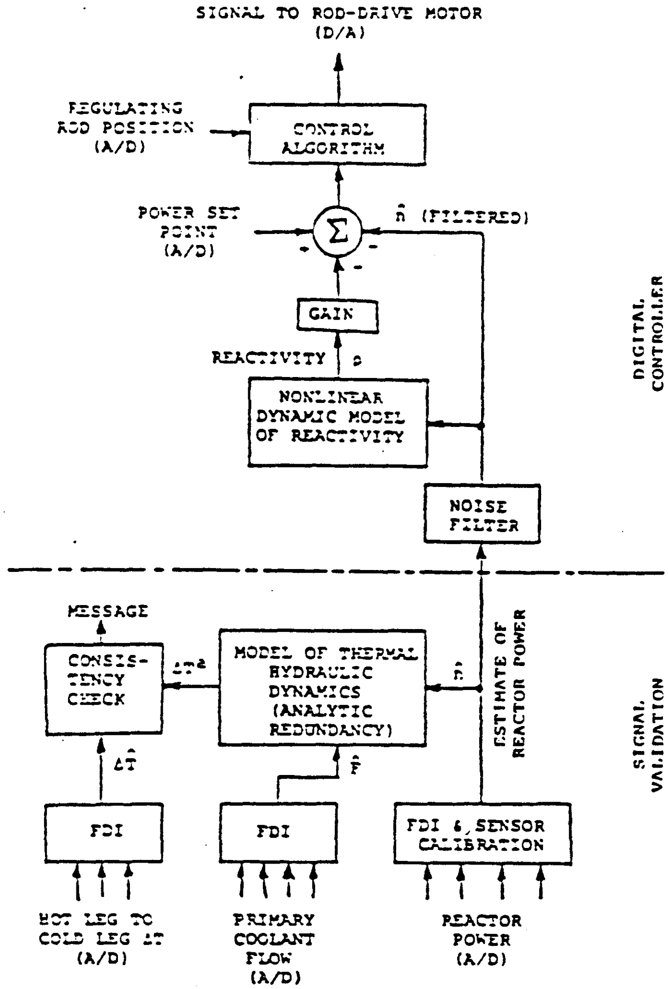

Figure 4. Signal Validation and Control System

because these sensors, which measure the leakage flux from the core, are spatially fixed. Any process that perturbs the leakage flux will cause the output of the neutron sensors to vary even though there may have been no net change in the overall reactor power. For example, as short-lived fission product poisons build into the core, the shim bank must be withdrawn to provide compensating reactivity. This sequence entails no change in reactor power but, as the shim bank is withdrawn, the magnitude of the peak in the axial neutron flux profile decreases and its position relative to the bottom of the core rises. Hence, the outputs of the neutron power sensors may change even though reactor power, as indicated by heat balances, is constant.

Given that process disturbances, such as xenon transients, coolant temperature variations, and movement of individual shim blades, can affect the neutron power sensors and since the reactor control system feeds back the validated power estimate at each sampling interval, the neutron power sensors are calibrated online and the estimate is obtained as a weighted average of all valid, calibrated power measurements.

The output of the calibration filter is protected against possible drifting by testing it against an analytic redundancy that is valid for both steady-state and transient operations. A real-time thermal-hydraulic model of the primary coolant system that accepts the validated measurements of power and coolant flow estimates as inputs is used to generate an analytically redundant value, ΔT_a, for the hot leg to cold leg temperature difference [7]. The value of ΔT_a is compared with an estimated value of the temperature difference, ΔT_e, obtained from the sensor outputs at every sample. An inconsistency of ΔT_a and ΔT_e implies possible errors in the power estimate since it is one of the inputs to the analytical relationship for computing ΔT_a. If this occurs, an alarm message is generated to notify the reactor operator. This procedure also guards against common-mode failures that can not usually be detected from either power or ΔT sensors alone.

The process and measurement noise in the power estimate is filtered digitally before that estimate is used as a feedback signal to the controller. (Refer to Figure 4). The filter reduces the random movements of the regulating rod drive motor around an equilibrium position. The digital filter, in its present form, is a 2-pole Butterworth. Its cut-off frequency is a selectable parameter having typical values in the range of 2π to 5π radians/sec.

In addition to feeding back the power estimate, the controller is routinely calculating and feeding back the reactivity deviations as shown in Figure 4. An inverse kinetics algorithm is used for on-line evaluation of the reactivity from the reactor's power level and history [10]. The advantage of reactivity feedback is that whenever the equilibrium critical condition is disturbed, the reactivity promptly changes thereby creating an appropriate control signal that will adjust the regulating rod's position to restore the original condition. Reactivity feedback provides a fast, anticipatory control action that is independent of the steady-state value of the reactor power.

The regulating rod's associated reactivity, ρ, is a nonlinear function of the rod position, x. Given that the reactor power is regulated by controlling the reactivity which, in turn, is influenced by the regulating rod's position, the overall plant gain is directly related to $d\rho/dx$. Hence, the plant gain varies nonlinearly with changes in the regulating rod's position. In order to compensate for these variations in gain that accrue from the nonlinear relationship between rod position and reactivity, the controller gain is constrained to be proportional to $dx/d\rho$. This is accomplished by approximating the regulating rod's differential reactivity work curve as a piecewise linear function and by measuring the rod's height at each sampling interval.

Safety Features

The digital control scheme of Figures 3 and 4 can be activated by one or more power sensors. It is not restricted to specific control structures such as P-I-D in the analog controller. Functionally, this control scheme is more flexible and, from the point of view of sensor degradation and failures, and the possible malfunctioning of other pertinent plant components (such as primary coolant pumps), more reliable than the original analog control scheme. However, since the computer system hardware, in its present form, consists of single-string components which are therefore not fault-tolerant, the experiments have been designed so that any possible hardware failures in the computer system will trip the controller to the manual mode as required by the MIT Reactor Safeguards Committee.

A transfer from "automatic" to "manual" control is initiated if any of the following events occur:

(1) The computer changes its state from "run" to "halt". This action provides protection against catastrophic hardware failures such as failure of the CPU and/or memory.

(2) The computer does not function in a cyclic fashion, i.e., the A/D, clock, and D/A subprograms are not periodically called. This action primarily guards against major malfunctions of A/D and clock modules, and any software failures that may lead to either termination or delayed execution of the program.

(3) The output of a particular power sensor exceeds the reference signal by a preset bound, typically 2% of full power. This action protects not only against hardware failures but also software failures and malfunctioning of the control law.

(4) Simultaneous multiple sensor failures, possibly due to a common cause. This means that a valid estimate of reactor power could not be generated using all available redundant information. This feature guards against an indecision.

(5) The reactor period is shorter than a conservatively set value. This action is a protection against an unforeseen situation that might drive the regulating rod out continuously.

Systematic procedures have been formulated for off-line and on-line testing of the system software. Therefore, protection features noted in items (1) and (2) might be waived if the single-string hardware of the computer system were replaced by redundant fault-tolerant hardware. The protection features noted in items (3) and (4) may not be necessary for less critical and more slowly varying processes such as chemical and fossil power plants where an alarm could be initiated instead of a trip, and the past value of the estimated variable be taken as a control signal. The protection feature noted in item (5) is specific to the particular application.

Results of the Experiment

The digital control system has been tested under both steady-state conditions and operational reactor transients and that are induced by both natural phenomena, such as xenon effects, and operator actions. A sampling interval ranging from 0.2 to 0.5 seconds was used for most of the tests. The observations from the experiments are as follows:

(1) The digital control scheme is tolerant of one or two failures in the power, flow and temperature sensors. This fact has been verified by both natural and induced sensor failures that have occurred during the test runs.

(2) The digital control scheme is capable of calibrating the power sensors on-line during transients involving xenon variations, fuel depletion, temperature fluctuations, and operator-induced power-demand changes.

(3) The digital controller maintains the reactor power at the desired level during both steady-state and transient conditions. It does this even though the neutron sensors themselves may be affected by process disturbances.

(4) The digital controller successfully maintains the reactor power at the desired value and is less sensitive to sensor degradation and noise than the original system.

(5) Controllers that feed back both the reactivity and the reactor power deviations are

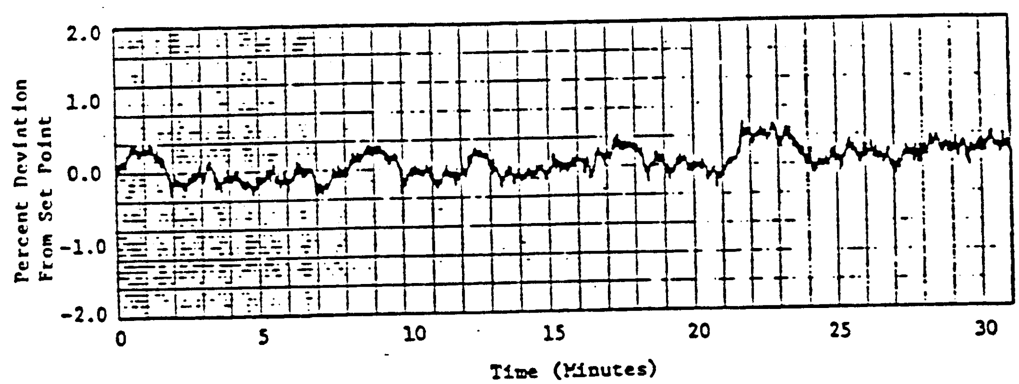

Figure 5(a). Steady-State Power Profile Under Analog Control

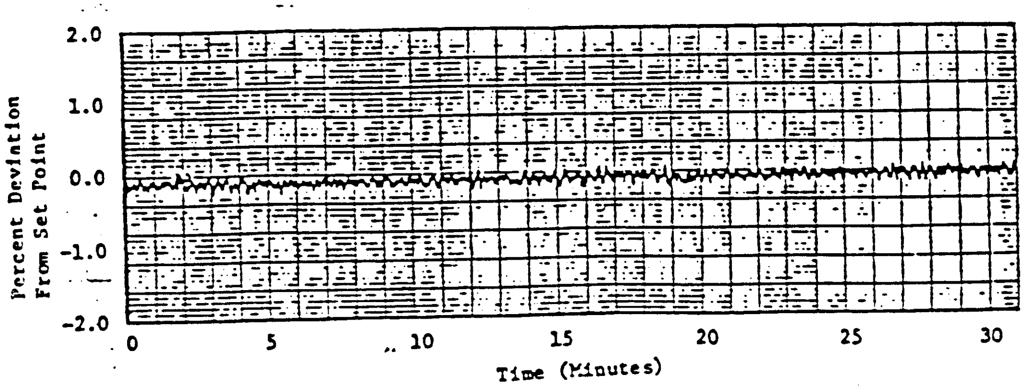

Figure 5(b). Steady-State Power Profile Under Digital Control superior to those that only feedback reactor power deviations.

(6) The system reliably displays the on-line information on sensor failures and estimates of measured variables, thereby assisting the operator in making timely and appropriate decisions.

Data taken during normal steady-state operation from one of the neutron flux detectors are shown in Figures 5(a) and 5(b) to illustrate the relative performances of the analog and digital control systems. The digital controller maintains the reactor power steady within a much narrower band than does the analog one. It appears from Figure 5(a) that the analog controller causes the reactor power to oscillate at a frequency lower than that of the reactor noise. These oscillations can be attributed to the relay logic in the analog control law which specifies both a dead band and hysterisis as shown in Figure 2. The tiny ripples observed in the power profile under digital control in Figure 5(b) are generated by inherent process disturbances. The continuous-action digital controller is capable of moving the regulating rod at a variable speed to compensate for part of the process noise within the pass band of the controller filter. In contrast to the fixed gain, reset rate, and the derivative time in the analog (P-I-D) controller, the digital controller has a variable gain which is automatically adjusted for the nonlinearities of the process gain. Also, power unbalances (due to process disturbances) are dynamically compensated by the feedback of reactivity which is calculated on-line via a nonlinear dynamic model of the physical process. Thus, reactivity functions as a nonlinear compensator in the digital system's control law. The major reasons for the improvements under digital control can be attributed to the following:

(1) The continuous-action control law replacing the relay logic.

(2) A nonlinear compensator (involving gain adjustment and reactivity) instead of the fixed parameter linear (P-I-D) controller.

(3) Sensor noise reduction in the computation of estimated power.

The single-input single-output controller, described above, is scheduled to be extended to a multivariable digital controller which, in addition to the reactor power, will regulate the average primary coolant temperature by automatic manipulation of the secondary coolant flow. Research efforts are directed towards the development of fault-tolerant control systems for power reactors using advanced analytical techniques [11,12].

Conclusions

A digital control system has been developed and demonstrated for on-line operations involving the closed loop control of power, fault diagnosis, sensor calibration, and information display on an NRC-licensed nuclear reactor. The control system software is structured for both systematic testing and implementation on commercially available microcomputers. Replacement of the existing single-string hardware by a fault-tolerant computer system such as FTMP [13] will lead to the evolution of an integrated control system that is completely tolerant of failures of the computer, sensors, and other plant equipment.

This technique may be reliably applied to large and complex industrial processes such as chemical and power plants for on-line fault diagnostics, sensor calibration, bumpless process control, and information display and monitoring.

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